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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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FUTURE PLANT DESIGNS SUBCOMMITTEE

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WEDNESDAY

MAY 1, 2019

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2D10, 11545 Rockville Pike, at 1:00 p.m., Dennis Bley, Chair, presiding.

COMMITTEE MEMBERS:

DENNIS BLEY, Chair

RONALD BALLINGER

MICHAEL CORRADINI

VESNA B. DIMITRIJEVIC

WALTER L. KIRCHNER

JOSE MARCH-LEUBA

JOY REMPE

PETER C. RICCARDELLA
GORDON R. SKILLMAN
MATTHEW W. SUNSERI

ACRS INVITED EXPERT:
DAVID PETTI

DESIGNATED FEDERAL OFFICIAL:
WEIDONG WANG

ALSO PRESENT:
DON ALGAMA, RES
STEVE BAJOREK, RES
JONATHAN BARR, RES
MICHAEL CASE, RES
AMY CUBBAGE, NRO
HOSSEIN ESMAILI, RES
BRANDON HAUGH, Kairos Power
DAVE HENDERSON, DOE
JOSEPH KELLY, RES
JOHN MONNINGER, NRO
BRAD REARDEN, ORNL
KIM WEBBER, RES

CONTENTS

P R O C E E D I N G S

12:58 P.M.

CHAIR BLEY: Good afternoon. The meeting will now come to order.

Folks out on the phone line, please keep your telephones muted. We're picking up some sounds from you.

This is a meeting of the Advisory Committee on Reactor Safeguards, Subcommittee on Future Plant Designs. I'm Dennis Bley, chairman of the Future Plant Designs Subcommittee.

ACRS members in attendance are Mike Corradini, Jose March-Leuba, Dick Skillman, and we have with us an invited expert, Dave Petti, over here. We export shortly for Member Rempe, Kirchner, and Sunseri and Dimitrijevic to join us later during the meeting.

Weidong Wang is the ACRS staff member who is the Designated Federal Official for this meeting.

The purpose of today's meeting is to review three draft reports prepared by the NRC staff on the status and plans for the development of computer codes to conduct confirmatory analysis on advance non-LWR reactor designs. It's actually two big reports and one little introduction report.

The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions as appropriate. At this time, this matter is not scheduled to be addressed at an ACRS full committee meeting.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, FACA. That means that the committee can only speak to its published letter reports. We hold meetings to gather information to support our deliberations.

Interested parties who wish to provide comments can contact our office requesting time after the Federal Register notice to the meeting is published. That said, we can also set aside for extemporaneous for members of the public attending or listening to our meeting. Written comments are also welcome.

The ACRS section of the U.S. NRC public website provides our charter, by-laws, letter reports, and full transcripts of all full and subcommittee meetings, including slides presented at the meetings.

Detailed proceedings for the conduct of ACRS meetings were previously published in the Federal Register on December 7th of 2018. The meeting is open to the public attendants and we have received one request to make an oral statement. That will be at the end of the meeting.

Time has been allotted in today's agenda to allow for this statement and for any other spontaneous statements from the public.

Today's meeting is being held with the telephone bridge line, allowing for participation of the public over the phone. A transcript of today's meeting is being kept and let me interrupt. We're having some technical difficulties again in this room. The connections over to the court reporter's station aren't fully working. Anybody who is at this front table will be well heard. People who have comments to make instead of coming to the microphones scattered around the room, please come up to the chairman of the ACRS's chair and sit down and make your comment here, because that's the only way it will get on the record.

We request that meeting participants on the bridge line when they are called upon to identify themselves when they speak and to speak with sufficient clarity and volume so that they can be readily heard. Same thing for up here. Participants in the meeting, do as I just said.

At this time I ask that attendees in the room please silence all of their cell phones and widgets. I remind speakers at the front table to turn on their microphone indicated by the illuminated green light when speaking, and turn it off when you're not speaking or we get noise on our system.

We will now proceed with the meeting and I'll call Kim Webber, Deputy Director of the Division of Systems Analysis, Office of Nuclear Regulatory Research to make introductory remarks.

Kim, please.

MS. WEBBER: Thank you, and thanks for inviting us to talk with you about our approach for evaluating, developing, and assessing codes that will support the staff's future licensing reviews of non-light water reactor or advance reactor designs.

You may recall we had a meeting back in August 2018 where representatives from the Department of Energy and also from the national labs came to present information about the DOE codes that are part of the Nuclear Energy Advanced Modeling and Simulation Program and they describe the co-capabilities in addition to the degree of validation which I think is a key interest that you have.

Also, back in November there was a meeting that was involving Office of New Reactors, John Monniger came to talk about the confirmatory analysis approach that NRO is thinking about for these licensing actions coming up and we also had representatives from the technology working groups come and talk about their advancements or their state of where they are with codes modeling and simulation.

And so today, we come to talk to you about where the NRC is and how long we have been a preparing for code development activities that will support future licensing.

So today, I'm here with Steve Bajorek and Joe Kelly. They're going to talk about design-basis event analysis and then after the break session, Hossein Esmaili, Don Algama, and Jonathan Barr will talk to you about source terms, severe accidents, and accident progression analysis.

I want to say that each of the presenters, when we had a pre-meeting with you, I think your primary interest as we understood them was to understand the important scenarios and phenomenon that are considered safety significant relative to the various advanced reactor designs, in addition to the gaps that we have in terms of our knowledge of the phenomenon. So that's what each of the other presenters is going to discuss in addition to considerations for selecting various codes which they'll talk about.

Before they get started, I wanted to give you just a brief overview of our approach to code development activities that are represented in the introduction report which is again the skinny one that Dennis referred to. And other speakers will talk about more details of these other two reports that we forwarded that are also publicly available.

The reports actually represent our current thinking and understanding of the wide range of advanced reactor designs that are out there. And so we don't -- as you can appreciate, we don't have a lot of design detail, details about designs at this point in time. And so you know, we expect that our plans will evolve as the information becomes available to us and you'll also hear that we've taken certain approaches to make efficient use of taxpayer dollars and government resources.

MEMBER REMPE: As I'm flipping through this report, I can remember a question that a former member asked about software and modeling simulation codes in another avenue a couple of years ago.

Does the Agency have a high-level goal of what they want this offer to be? Like do they want state of the art, state of the practice? Is that goal emphasized in your philosophy at a high level here?

MS. WEBBER: You know, I'm not sure. Let me try to take a stab at it, but I'll look to Steve. I mean at the present time, the designs are evolving and so we're trying to ensure that our codes are ready to the best we can to evaluate behavior under various actions and operating condition scenarios, so I would consider that we're trying to become state of practice as information becomes available.

MEMBER REMPE: That was the response years ago and I just wanted to make sure that's still what the Agency desires, state of the practice, not state of the art.

MS. WEBBER: And I'll turn to Steve to see if he has any other qualifiers.

MR. BAJOREK: Yes, I think that's pretty accurate. The guidance we've usually followed is we aren't supposed to be leading the state of the art, but we need to make sure our tools are at least consistent with those of the applicants and they're sufficient so that we can make our regulatory decisions. But we're not out there to try to take the technology and advance it by ourselves, but we have to stay with it and that's always been a moving target.

MEMBER REMPE: Thank you.

CHAIR BLEY: Steve, before you leave that and you don't have to answer this now, you can answer it a little later, but for the codes you guys are talking about, this predictive capability maturity model matrix to evaluate maturity levels, I don't think I saw anywhere where you relate maturity levels to the kind of things Joy just asked you.

Where does the state of the practice live for each of the elements versus maturity levels? And how do you decide if you're there or how far do you have to go?

MR. BAJOREK: That's a really good question and if you look at Bill Overcamp's description of the PCMM which you're referring to --

CHAIR BLEY: Yes, I've read that.

MR. BAJOREK: Page one or two basically says it is not really something that tells you if it's sufficient. That has to be up to those who are using it.

CHAIR BLEY: Or those who are reviewing it.

MR. BAJOREK: Or those who are reviewing it. It's a good point.

I think what the staff is going to look at to PCMM for is to get it to the point where the codes are good, but not necessarily at that highest level because we often take some of the models that we have to develop and we'll simplify them and we'll bound them. That's good enough for safety. So that would allow us in that PCMM chart to stay with something like a two as opposed to going up with three which the applicant may want to have if they're going to optimize their design.

CHAIR BLEY: Another thing that will get asked about later, but I'm going to -- don't answer it now, wait until you get there later. If you go through for a whole bunch of codes and evaluate them against that matrix and say where things live and some of the codes almost always live down at the zero or one level and is that good enough in some cases? When you get there, talk about that a little bit because it's -- what you just talked about, I don't find them in your report.

MR. BAJOREK: Okay.

MS. WEBBER: Okay, and so just moving on, so as you all can appreciate code development activities are time consuming and somewhat resource intensive. But we believe that it's a very important activity to help build staff expertise when it comes to being able to conduct licensing-type activities.

In addition to utilizing the codes for the initial licensing application reviews, these codes can also be used for confirmatory analysis to support future reactor design changes and also to develop technical bases to support rulemakings or specialized safety studies. So it's not just that we're developing these codes for the initial application review.

Over the last few years, NRC's code development activities have been modest, in part, due to the uncertainty in priorities in the non-light water reactor industry for licensing a wide range of technologies. So what you see here on this slide there are essentially 20 advanced reactor developers and as represented by the blue and white box shown in the lower left corner, there are basically five companies who have actually given us information that describes their regulatory engagement strategies.

So out of the 20, there's 5 who have come and spoke to us about their plan. So it's a very uncertain environment for us and in terms of utilizing resources, we need to remain flexible as industries plan to change. And I think our code development plans reflect this.

MEMBER CORRADINI: So what is a RIS

response?

MS. WEBBER: Regulatory Information Summary.

MEMBER CORRADINI: So they responded to your request?

MS. WEBBER: Right, and I don't know if you want more details about that particular RIS, but we have Amy Cubbage in the audience who can talk about that.

MEMBER CORRADINI: You can go to the chairman's chair.

MS. CUBBAGE: Hi. Amy Cubbage, Office of New Reactors. So we periodically issue a generic communication called the Regulatory Issue Summary and we request that any potential applicants give us information so that we can inform our budgeting process. And so these companies highlighted have responded to the Regulatory Issue Summary and indicated that they would like to engage in either pre-application or ultimate --

MEMBER CORRADINI: Okay, so it's basically their intent.

MS. CUBBAGE: It's an intent to engage with the NRC.

MEMBER CORRADINI: They may or may not

have as of yet, but that's their eventual intent.

MS. CUBBAGE: Exactly. So for those companies listed on the slide, we've initiated formal pre-application engagement with X-Energy, Kairos, and Oklo. And we've had some informal discussions with Terrestrial and TerraPower with the anticipation that we would engage in pre-application or actions eventually with the others. And we are expecting that we will get other RIS responses, but it is a very uncertain environment as Kim mentioned.

MEMBER CORRADINI: Okay, fine. Thank you.

MEMBER REMPE: So before you leave that slide, in follow up to my question earlier and your response that you want to have modeling and simulation tools that are commensurate with what is being used by these design developers, the ones that have responded to you or in your interactions with them, what kinds of modeling and simulation tools are they planning to use?

MS. WEBBER: I think some of that information is not publicly available at this time and --

MEMBER CORRADINI: I think in the November 16th meeting we did get that current summary.

MEMBER REMPE: We did. So maybe I should

rephrase this as we're on the record here. What level, what sophistication are you seeing coming from these design developers?

MS. CUBBAGE: So maybe I would just like to clarify that the Office of New Reactors, we lead the pre-application interactions and we don't have any more information regarding which codes and which level of sophisticated they're at than what you've heard in November. And then the Office of Research is working on our capabilities.

And to your point earlier about do we need to be on par with them to, in general, for us our choice to do confirmatory analysis, that's our choice whether we do it or not. We may not do it in all areas. So the applicants are going to be responsible for having analysis across the board to the appropriate level to support licensing. We're not held to that same standard. We may pick and choose where we want to do confirmatory analysis in the areas of uncertainty or where we want to explore margins.

MR. BAJOREK: I think just to maybe get at your question, there's actually a range of codes and capabilities that they're looking into. Some I would think we would consider very close to state of the art, fairly advanced. Some, because they have access to codes that have been used for that type of design for many years, are staying with more of the Legacy-type tools. So it's a variation from more conventional-type codes all the way up to some that might be state of the art.

MEMBER REMPE: Well, I have to go back and look at my notes again. Some were, as you are, considering the DOE codes, but the issue, of course, was ownership of the software and they have not figured that out and possibly because they didn't have the resources to perhaps -- or access to Legacy code.

I guess I left that meeting thinking that they were not generally relying on very sophisticated state of the art tools. And that's why I'm trying to bring that up again.

MS. WEBBER: It is their choice, their choice. And I think as you'll see in a few minutes, we have chosen the approaches that we have taken for a lot of reasons including trying to be resourceful given the uncertainty in the environment that we're at which is a good segue to the next slide.

So in speaking about the environment and the uncertainties surrounding where the industry is going, we really need to be efficient with our resources and so I want to briefly discuss this aspect.

So as the NRC funding decisions occur, the staff will need to be conscientious and deliberate about which tasks receive funding since not all tasks in our plans can be funded in any given year. Furthermore, as our code development plans evolve or they will evolve in response to changes with industry plans and as NRC budget considerations occur.

So the staff code development plans are comprised of approaches and tasks utilizing the following considerations. That is we're prioritizing tasks that support flexible codes that can be used for a wide variety of reactor technologies. We're initiating tasks that require long lead time data needs such as the molten salt reactor properties. We want to prioritize when we can promoting staff's understanding of a potential applicant's codes and methods. And as Joy pointed out, their plants are evolving and so as we understand what codes they're going to use we'll pay more attention and spend more time understanding those codes.

We also want to use our resource at least at this time to identify database needs or data needs that apply to multiple designs rather than data that could be used for a particular design type. And then obviously, we'll minimize costs by leveraging DOE tools and coordinating code development plans with them.

And so for some of these reasons that we are pursuing different approaches for design basis accident and beyond design basis accident analysis. And what I mean by that is that for DBA, we believe that the use of DOE developed codes through the NEAMS Program will help augment the NRC's non-light water reactor computational capability in a cost-effective manner. Whereas, for beyond design basis analysis tools, the use of MELCORE, SCALE, AND MACCS seems to be more prudent because basically those codes are -- in the case of MELCORE and MACCS the only severe accident source for accident progression codes that seem to be able to provide the capabilities that we need.

CHAIR BLEY: Kim, before you go forward, you talk a lot about the uncertainty and all the different things that might come before you which makes sense to me, but in Section 2 of your report on licensing basis events, let me read you a sentence and maybe you can explain it to me. In order to be ready for confirmatory analysis with an aggressive schedule such as this, and you gave one, it is important that the codes selected for design basis event analysis be essentially complete so the code validation can begin in 2019.

Given all the uncertainties you talked about, you still mean that? Or is that just one or two things that you expect to be validating.

MS. WEBBER: No, I mean I think clearly our code development plans, we have very strong code development plans I think are representative in these reports that reflect the kind of knowledge that we have today. And I think it's a very solid plan and a solid approach.

I think that as we get an understanding of where the industry is going, we may need to change our strategy and the points on this slide may be different. If there's a front runner who we know is actively engaging with our regulatory office, we may decide that we need to focus our energy and our resources towards being prepared to support licensing in that particular design.

So we're trying to be flexible. We're trying to be resource efficient, but we're also actively monitoring and engaging with the technology working groups and external stakeholders to understand what their plans are.

MEMBER CORRADINI: So let me ask a different question. I'm not sure where Dennis was reading, but I was reading in the shorter document.

MS. WEBBER: Yes.

MEMBER CORRADINI: You have efficient use of resources and you go through a number of things on page on section 23.

I guess my question kind of goes like this. Is it -- and I didn't see it, so maybe I'm missing it. You just tell me where to look. Is validation part of the plan and if validation -- what I'm thinking is experimental validation is part of the plan and if experimental validation is part of the plan, whose basket is it in? Is it in the industry's basket? Is it in DOE's basket in support of the industry or is it NRC's? Because it seems to me the common element here that is -- I mean I wrote down attributes for tool selection and one of them was validation, right? And so I'm kind of curious about that as to -- because that, to me, I think I'm speaking not just for myself, but for a number of the members, that's a big one.

MS. WEBBER: So the short answer is yes and Steve is going to give a much better answer than

I could.

MEMBER CORRADINI: We can wait. We can wait.

MEMBER REMPE: Yes, to what? Whose basket is it in?

MS. WEBBER: Yes, that validation is a key part of our code development plan.

MEMBER REMPE: So I didn't hear the answer to the question.

MR. BAJOREK: I'm going to talk about that more in my slides, but I think the answer is we're going to work jointly with Department of Energy to get the validation complete.

MEMBER CORRADINI: Okay. So then, I'm just kind of going down my personal list. You have in your first bullet and it was referenced somewhere in this report and I can't find it now. There was an indication there was a gap analysis for DBAs and severe accidents. Where is that documented?

I'm looking for a gap analysis that says tool X is okay, except for this gap. And we're going to fill this gap by doing step Z. Where is that documented?

MS. WEBBER: So I think in this particular report what I was referencing was a -- I believe it was a 2018 progress report that we wrote.

MEMBER CORRADINI: Okay.

MS. WEBBER: And we did some preliminary assessment of where we thought the gaps were so that we could then in out years focus our resources.

So what we you see in front of you is the result of the staff's assessment of where the gaps were. And I think the reports themselves describe where the current gaps are.

MR. BAJOREK: Yes. The intent of these, both the Volume 1 and Volume 3 and the other volumes that you'll eventually get are to identify the gaps. At least in Volume 1, I'll talk about that in a few minutes.

MEMBER CORRADINI: Okay. Fine.

DR. PETTI: Can I ask a question? I know you've only started looking at these gaps. If you have to say today based on what you've done so far, how much of the gaps are in the fundamentals of physics and chemistry versus some structure of a tool itself where I need a capability that's not rocket science? I don't have to go measure something, I just have to have the capability. Because in my mind, those are two separate things and really important. Where do you see most of that?

MR. BAJOREK: I think we're going to get two different answers. One for design basis type tools, where we think the functionality is there. We think most of the physics is there. You're never sure of that until you complete all that validation.

As we start to look in design basis, excuse me, beyond design basis events, some of the basic physics still needs a lot of experimental work to be done in order to fill in those gaps. There is some of that in design basis. I do have a slide that kind of addresses that, but I think most of the place where we need more experimental work where we're filling in those types of gaps lies in the severe accident arena.

MEMBER REMPE: Before you leave this slide, I had a couple of questions. One, with this coordinating development plan, will NRC have some way to have ownership of the source coding if DOE for some reason gets abolished. That was discussed. Or something like. What's your insurance by coordinating with them that you're wisely investing your resources?

And then secondly, in the LMP, and I've seen some of your documents, but why between design basis and beyond design basis is going to be fuzzy and in particular I think of gas reactor stuff because of my background. Circulating activity is what dominates the source term for a lot of AOOs and DBEs. And yet, the design basis event codes could kind of say now we're going to rely on MELCORE for that and so -- and historically, when I do an analysis you want to know that your sever accidents correctly predicting the thermal hydraulics and the source term, during early stages and before you get to high temperatures in the gas reactor, for example, where you have additional releases.

And so coordination is another question with the advanced reactors that are going to be so much safer with minimal releases that I think circulating activity in releases is the gap.

MS. WEBBER: On the latter question, Steve has got a slide specific to that question.

MEMBER REMPE: I'm interested in hearing it.

MS. WEBBER: And on the first question I think you were asking about coordinating development plans and who is going to own the source code. I think we've had some discussions, but I don't think there's been any decision about that yet.

MR. BAJOREK: I think that's still down the road for us to decide, but our goal is, at least for design basis tools, we have the source code. Actually, we're running -- we compile the source code now. We have access to that. We will continue --

MEMBER REMPE: That's good to hear. I had not heard that previously, so that's good to know.

MR. BAJOREK: We'll talk about that in my presentation.

MEMBER REMPE: Thank you.

MEMBER MARCH-LEUBA: You not only have to have the source code, you have to have the copyright or the right to use it. If the code gets transferred to another company you lose it. We have to think about it.

CHAIR BLEY: I want to come back to what I started. I had rather hoped on my question of validation this year that you would have said something like well, we're ready. And if I look at your matrices on maturity levels for the HTGR, we're ready this year and we're ready to go. For the gas-cooled fast reactor and molten salt, now we're way down the line. You didn't say that. Is that what you meant?

MS. WEBBER: Steve is the king of PCMM.

CHAIR BLEY: What I hear Dennis asking is there are certain things that are mature and good enough, but not the best, but good enough. And there are certain things that aren't good enough. And it's not a PCCM thing. It falls out into the PCCM if you do it right.

MR. BAJOREK: And I think you will hear a recurring theme from both me and Hossein and other speakers, that for gas-cooled reactors, our experience with NGNP has put us in pretty good shape.

Sodium fast reactors, there's been enough work. As we start to go to the micro reactors and certainly the multi-salt, the fuel salt, they get more and more mysterious and the path forward is more difficult and it's going to take us longer to get there. So I think you're right. Gas-cooled reactors when it comes to confirmatory analysis, we feel pretty good about that. But if you go to some of those other designs, that's going to take a little bit longer.

And hopefully, that is reflected in the PCMM.

CHAIR BLEY: I think it is reflected in the PCMM and I think you maybe could have done a little more of this kind of high-level implications of what you found when you did that analysis.

MS. WEBBER: I think that's a good suggestion. I have two more slides left. I want to talk about some of the more recent accomplishments. So in FY17, NRC started receiving off fee-based funding to support broad range of strategies including the NRC's implementation action plan Strategy 2 which is for code development activities.

Today, most of the resources have been used on these activities that are listed on the slide. We covered conducting preliminary gap analyses to identify where to place our resources.

We've been working very closely as it's come up already with DOE and the national laboratories to identify and evaluate the capabilities of the CASL and NEAMS codes for NRC's use in the design basis in performance -- fuel performance and design basis analyses.

And staff has been receiving training on the various reactor technologies. I think it was maybe 18 months ago or 2 years ago we received training on general molten salt reactor technology. More recently, we received training on sodium fast reactor technology and then I think upcoming this summer we have training on high temperature gas reactors.

Additionally, DOE and the national laboratories have bent over backwards to come help us understand how to use their codes with hands-on training here at the NRC offices. And we've also sent staff like Joe Kelly out to the Idaho National Lab to learn how to use those codes.

Additionally, we've performed some preliminary code development activities in the field performance area and accident progression area and you'll hear a little bit about the accident progression area when Jon Barr comes to the table. And then also a large amount of the resources have been spent developing these plans today.

MEMBER CORRADINI: So I have one more attribute that's getting to me. I wrote down to myself minimal set. In other words, never do more than you need to do. And I'm trying to understand how the activity identifies what's the minimal set or to put it a different way, this is not the right answer, but this is a possible way, the NEAMS codes are my impression. I could way off base. The NEAMS codes are quite mechanistic in comparison to past, but that mechanistic attributes may not be -- may not contribute much to the uncertainty.

Has there been any sort of thought about what is the minimal set? If I do uncertainty analysis with the codes, I find out that these five things are interesting from a physics standpoint, but make no difference to the results so that I've used the detail tools that help drive a simpler tool in evaluation.

MS. WEBBER: So I think we started having those discussions amongst the staff in terms of the complexity of the reactor designs and what code is more informative to use to evaluate those kinds of things. But I'm not sure.

So we've also talked about conducting specialized studies, reactor technology sort of studies to figure out where to focus in terms of the most uncertain areas, if you will. And so I think that still is evolving amongst the staff, but I'll ask Steve if he wants to contribute anything.

MR. BAJOREK: Yes, that's an important point and as we've been evaluating the codes, one of the things that we want to be able to do is simplify things that we use. We don't need to get a pellet by pellet power distribution as you've seen in some of the CASL studies that we've done. They do a marvelous job. And those types of things might be very essential for a core designer who wants to minimize the number of feed assemblies or even to optimize performance. But as we're looking to margins to various safety limits, we don't need that detail.

So as we've been setting up and evaluating some of the models right now, one of the things that we're trying to do is to see can we take these codes, simplify the nodalization that speeds them up, makes them more flexible so that we can instead of going after very high amount of detail look at the broad number of accident scenarios, faster, not as sophisticated nodalization, but that allows us to look at a lot of different scenarios and get to that safety question which is different from what the applicants use.

MEMBER MARCH-LEUBA: I would like to second that. We are all code guys. We're all physicists. We like details. We love the details. But all these new reactors have humongous margins. You can considerably use a steady-state stimulator between point A and point B and bound anything that can happen in between. So yes, a judicial use of uncertainty in my calculation to bound the result, I say yes. That will save us a lot money, plus give you the confidence. When you went to the fine mesh and fine detail, you may get a more accurate result. You may get the completely wrong result because you just go into too much detail in the code. And when you use the high level, I totally support that.

MS. WEBBER: When we get to Steve's presentation, he'll talk about reference plant models and being able to do some of those kind of assessments.

MEMBER REMPE: I had one other question, too, about completeness. I know you did PIRTs to try and emphasize what phenomena were important, but I didn't see how complete the PIRTS were. Did you look at a range of conditions, for example, if you're going to transport a fully loaded reactor and install it somewhere and then remove it. Maybe transportation is where it happens. We've learned with another designer that moving a fully loaded module and dropping it might be a high-risk dominant event. And so I'm just wondering if we're not -- we've got the proper focus on where the high-risk events will be.

MS. WEBBER: So I think, you know, clearly that's something that we need to look at and consider in terms of the events. And so jumping to this next slide, part of our plan is to not only address sort of reactor phenomenon, but also to look at radiation protection, siting analysis, dose calculations, and then also we do want to address any kind of computational needs and approaches for the back end of the fuel cycle including transportation for sure. We just haven't embarked on that yet.

And so again, we've received a modest amount of resources, so we haven't thought about all of those considerations, but they are clearly important areas that we need to look at. It is in our thinking that we need to look at those kind of activities. And so we will be looking at them.

CHAIR BLEY: You know, the things Mike and Jose and Joy have just brought up, to me are kind of first order things that decide how far you need to dig into some of these areas.

Just off the top it feels like you're looking under the street lights for the keys because you like the details. Me, too. But maybe we're not organizing our resources in the most effective way.

MS. WEBBER: That's a good thought that we'll have to take back with us, I think.

So let me get to the last slide, so I can move on to Steve and Joe's presentation. So we've been preparing our plans. They're currently represented in these three volumes, introduction. It is essentially an overview to our approach to code development activities.

Volume 1 focuses on design basis accident analysis tools. And then Volume 3 focuses on source term analysis, severe accident progression, and consequence analysis,

Our fuel performance volume is under development in addition to a Volume 4 which we're thinking is going to contain radiation protection and siting review, dose assessment codes. And also in that context, we're trying to consolidate codes where we can, so in the context of Volume 4, there are a lot of atmospheric transport models that are in same codes. So we're trying to see if we can't come up with a more efficient way to address those potential licensing needs.

CHAIR BLEY: Are Volume 2 and 4 --

MS. WEBBER: A work in progress.

CHAIR BLEY: Pretty far along or are they just starting or --

MS. WEBBER: They've got a lot of work to do. I mean I should say that the code, that Volume 4 is probably further along than the fuel performance one, but we actually have had the loss of a significant staff person who's going to work in industry and so we need to reconstitute the staffer developing --

CHAIR BLEY: So we won't see them for some time.

MS. WEBBER: It will be -- yes, a few months.

CHAIR BLEY: Months, not years?

MS. WEBBER: No, months.

CHAIR BLEY: Okay.

DR. PETTI: Just a clarification, a fuel volume is really DBA ALO focused?

MS. WEBBER: Yes, I think, transient focus.

MR. BAJOREK: It will serve a couple of different things. Fuel performance in terms of getting the design basis code during initial condition, but also dealing with the types of phenomena that may go on during transient or during severe accident of a transient nature.

DR. PETTI: It's just that fuels leads into Volume 3, right? As I'm reading it, I need that piece to understand, there's a leakage there, right?

MS. WEBBER: Yes, we understand that there's a connection. We've had many discussions, right? We've had many discussions. So each of the volumes that has a description, the codes and their usage, their regulatory applications, the rationale for selecting the codes and then the maturity which we've discussed a little bit.

It does identify information gaps and the tasks that we think need to fill those gaps as we understand the gaps today. I do appreciate the comment about maybe we need to look at transportation and other kinds of accident scenarios that are outside of the realm of sort of the large light-water reactor sort of thinking.

But I think as the details become -- more details become known to us about the reactor designs and the direction the industry is going, our plans will change and evolve. And you know, we may decide, I kind of alluded to this earlier that it may be more optimal to support licensing by doing some kind of a safety study, rather than doing specific confirmatory analysis or maybe a combination of both depending upon on the complexity and design and the information that we have at hand.

So with that, I'd like to turn over the presentation to Steve Bajorek.

CHAIR BLEY: Not so fast. We have Volume 1 and Volume 3. And they're both produced by the staff.

MS. WEBBER: Correct.

CHAIR BLEY: With the same goal in mind,

identifying the codes, maturity, the gaps, the technical gaps. To me they read as if two different worlds wrote them. The approach to identifying gaps is different, the approach to identifying maturity is decidedly different. Is there a reason they're different? Or each of you can tell us as you're doing it why you did it the way you did it.

MS. WEBBER: I mean so from my perspective with Steve and all staff who are presenting today. The codes have come up over the years and been developed in different ways, so there's a different perspective on what the gaps are and how to fill those gaps. And so I'm not surprised that you see that difference.

Maybe we could take a more deliberate approach to identifying gaps and filling the gaps. But I think I'd rather let the staff talk to you about how they went about doing that.

MEMBER REMPE: But to follow up on Dennis' question, in the future, I'd like to strongly --- that's why members comment, recommend that the staff that did Volume 1 and 3, look at what they've produced and some things like just experiments for data, that can be used to validate, differ and completeness and try and resolve some of those differences.

MS. WEBBER: Sure.

MEMBER MARCH-LEUBA: One change to Volume 2, this is a little bit into the future, if you use the old standard, NRC paradigm, you can do all the analysis you want on new fuel, unless you focus on foreign TAs and burn them, you're not getting any indicators (phonetic).

So this is a point in which we can think in the future and use this advanced DOE code, going to extreme detail, not to replace experimental, but to minimize the number of experimental procedures you can use. I hope you are thinking about it.

MS. WEBBER: Yes.

MEMBER MARCH-LEUBA: And it's represented in Viking, so they can take --- they can take credit for it.

MS. WEBBER: I think that is something that we're very open to.

MEMBER MARCH-LEUBA: But it helps if the staff provides down in the front. That would be an acceptable path forward.

MS. WEBBER: Okay. I'm kind of surprised it's not there, but --

MEMBER MARCH-LEUBA: I'm surprised too.

MS. WEBBER: Okay.

MEMBER MARCH-LEUBA: This doesn't exist. But the industry has been risk averse.

MS. WEBBER: Yes.

MEMBER MARCH-LEUBA: If they know you are going to accept it, if you think it's a viable alternative, it would be very good to say.

MS. WEBBER: Thank you.

MEMBER MARCH-LEUBA: I certainly do.

MS. WEBBER: Okay, thanks. Any other questions?

MEMBER CORRADINI: Yes, just one. I'm into cartoon pictures. The NGNP 2011 process evaluation model process calculation is referred to in Volume 3. Is it not the same thing you'd want to do in Volume 1? I'm back to Dennis about having commonality in terms of approach. You know what I'm thinking about?

MS. WEBBER: You're talking about the figure that's elongated that shows when code nodality --

MEMBER CORRADINI: It basically says I do a steady-state analysis to provide me this, which tells me to do this, which tells me to do this, and that's from the thing that we saw back in 2011 for the NGNP.

It seems to me there's got to be -- I hate to use the word universal, but it seems to me there's got to like a universal evaluation model process and then depending upon whether I'm going down to low probability, high consequence events versus higher probability, lower consequence events I would tend to maybe branch, but at least I start off with the same calculational thinking process.

And I think Volume 3 just stole it straight away from the NGNP document. I'm pretty sure. At least that's the reference. Figure 1.1 of Volume 3. And it just struck me as -- for me, not knowing all the pieces, it struck me as you've got to do this and then you've got to do this and then you've got to do this. This is how they kind of connect together.

MS. WEBBER: So was there a question in that?

MEMBER CORRADINI: No. Just an observation. I'm kind of just trying to get back to Dennis in that you want commonality in how you want to attack this because in some sense the LMP implies commonality. The only difference is things that are low and high probability.

MS. WEBBER: Sure. Okay. I'll turn the

presentation over to Steve. I'll set you up there.

MR. BAJOREK: Okay, thank you, Kim. And good afternoon, everyone. My name is Steve Bajorek. I'm joined with Joe Kelly. We're both in the Office of Research and what we're going to talk about next is our approach for design basis modeling and modeling in simulation.

Just to make sure we are talking some commonality we understand that Chapter 15 versus Chapter 19 is really not going to be the way of doing it for non-light water reactors. This is going to be replaced with a Licensing Modernization Project where PRA insights and the applicants are eventually going to come to us and describe what are going to be the scenarios of interest.

But with respect for the use of these codes, whether we call them design basis or beyond design basis, the idea remains the same as we have done in the past and that the design basis tool will be those which will be used for confirmatory analysis of events where there's little or no fission product release and transport, where the core at the end of the day is still pretty much looks like the core that started at the beginning of the transient.

The design beyond design basis tools for those events where we see core disruption, core melt, fission product release and transport. The difference now with non-LWRs versus light water reactors is that design basis codes will likely to be used to look at events that will have multiple failures. They'll have conditions that may have been defined as beyond design basis back in the light water reactors analysis, unprotected loss of flow, loss of several heat sinks, multiple failures.

The Volume 1 that we're going to talk about it's going to focus on the design basis tools. As Kim mentioned, our approach has been to follow Reg Guide 1.203 and basically what we're going to be looking at phenomena identification, ranking tables, what do the phenomena that's going to be important, of interest, what's different about these non-LWRs that we need to have to account for that can't be accounted for in the light water reactors tools.

In each of the sections, we tried to identify the gaps which are remaining and those gaps are usually in the modeling capabilities, the validation, and in some cases, the available data. And finally, within having a plant model that represents what we think the applicant is going to come in with so that we can begin the process of identifying what are those margins and what are those processes that occur in the plant that may have the largest uncertainty that we have to address as we go further into this.

In Volume 1, we try to identify specific tasks that will get us to that starting line so these code or codes are ready for that confirmatory analysis.

MEMBER REMPE: Steve, talk a little about the PIRTs. I saw the names. I didn't look at the documents, but it looked like you had some lab folks. You were involved. I didn't see any names that struck home from design developers and I was thinking about again my comment earlier about looking beyond -- and maybe design developers wouldn't cough up such information. And I don't know a lot about these designs, but heat pipe reactors, to me, if you've got something with -- something fully through it, you'd want to look at their potential failures and collapse. I didn't see things like that just played out and lift off for gas reactor. To me, those would be releases that you would have in a design basis. And I know earlier you said you were going to talk about this later, but I read up and basically you're saying we're just going to use what comes with MELCORE is what I got for your conclusion. And I'm just kind of wondering why you made that decision.

MR. BAJOREK: Well, this is basically the same as what we've been planning to do with NGNP where the design basis tools would be looking at more of the pressurized loss of force cooling and the depressurized loss of force cooling, things that involved graphite dust, water ingress or air ingress because they're more chemical and aerosol in nature. Those are the types of accident scenarios that would be handled with the MELCORE.

DR. PETTI: So Joe, just to be clear, having sat on the NGNP PIRT, industry was not allowed to participate.

MEMBER REMPE: So you could not bring in any --

DR. PETTI: They could sit, but there was an official team that was probably the PIRT team. They could sit in the audience and they could help try to steer, but none of them were official members of the --

MEMBER REMPE: But they could provide comments.

DR. PETTI: They could provide comments.

MEMBER REMPE: And say hey you're missing the boat --

DR. PETTI: There were many. The other one like this print out in Lookout -- that's all covered in the fission product transport PIRT, not in the fuel PIRT that was done.

MEMBER REMPE: But again, if you have a -- back in the GA days, if you have a break, a pipe break, you depressurize, what's going to be source term? It's the circulating activity. And so --

DR. PETTI: It's all there.

MEMBER REMPE: Again, you're just -- how are you going to compare?

MR. BAJOREK: And that will be the case. When you have that circulating activity, the idea would be to have a code like MELCORE track it through the containment or confinement and then possibly pass that along to MACCS to look at the doses outside the plant.

For the design basis tool, your real concern are temperatures that you've occurred with on the fuel and on the vessel. Do you have any additional failures, okay, that would have any additional threats to a defense-in-depth.

DR. PETTI: You won't use MELCORE in the design basis events, correct?

MR. BAJOREK: Right.

DR. PETTI: What Joy is saying and she's right is that are some design basis events that will allow releases from the primary system. What tool will take it from the primary system to the site boundary?

MR. BAJOREK: And that would be MELCORE.

DR. PETTI: No, if it's a design basis event, you will use MELCORE for design basis events. Is that what you're saying?

MR. BAJOREK: You'd need it for transport.

DR. PETTI: You use it for -- okay.

MEMBER REMPE: What will you use these design basis codes for, just to verify that the vendor does fuel loading correctly?

MR. BAJOREK: No, because you're going to have to understand whether your plenum temperatures are so large that it fails to control rods, whether your vessel temperatures are going to be high that you'll have a vessel failure.

MEMBER REMPE: Can you not get that from MELCORE?

MR. BAJOREK: I don't know.

MEMBER CORRADINI: I was waiting for someone to say margin. It would seem to me the more sophisticated codes, assuming I want sophistication provide me margin to the failure point whereas if were to use something MELCORE which would be pardon the expression, cruder. I'm looking at the MELCORE people who are probably losing their minds over there, but it would be a little more integral in terms of behavior of the bounding.

MEMBER REMPE: So then you get back to my question about how much state of the art versus state of the practice and how much does the regulator need and that's a question that we'll have to all think about.

CHAIR BLEY: Amy?

MS. CUBBAGE: I just wanted to add don't get too hung up on -- this is Amy Cubbage again. Don't get too hung up on the DBE versus BDBE distinction. Focus on the types of codes and the phenomena that they're capable of modeling. I think we use some Legacy terminology in naming these particular reports for convenience dividing between different groups in the NRC. It's not that they'll be --- once you get into licensing modernization, you have to -- you're exactly right, the consequence, is it transport, is it -- it could happen in a design basis event and you may have no fission product or consequence and a beyond design basis event, depending on the frequency and the scenario.

MR. BAJOREK: And we think that, as we go on, there's probably going to be overlap in the capabilities in order to -- take a look at the different scenarios we have.

We've seen this slide before. Our approach to taking all of these various applicants and keeping some sanity about this so that we reduce the number of codes involved was to take these and define these in terms of ten different design types, each one with individual gaps and each one you may have to address in a different manner. This ranges from the high temperature gas reactor prismatic all the way down through your heat pipe reactors to your molten chloride salt reactors. Each one has its own types of gaps and we need to get the codes to be able to attack any of these types of designs.

MEMBER KIRCHER: I would just be careful if I were to go back to your first Slide 2. You are looking at some concepts that actually have circulating fuel. It seems to me that is a much bigger break point for you in your code selection and rankings and such than anything else that we've seen before. It just seemed to me that there would be a dotted line somewhere that says these concepts of circulating fuel that interjects a whole higher level degree of difficulty in terms of regulatory analysis and compliance determinations, so I would just interject that caution because that, at least I think from the technical standpoint, not from a personal opinion standpoint is a major demarcation, amongst that long list of concepts.

MR. BAJOREK: I agree and I think we'll see this. As I mentioned, we've been following our Reg Guide 1.203 code development process where we go through the scenarios, the PIRTs. We then look at the codes. And as part of this, we review those available PIRTs that have been available for all of these different things, from NGNP all the way through the heat pipe reactors.

I'll jump ahead a slide here. This is the list of PIRTs that we have primarily relied upon. Some of them started with NGNP. Molten chloride salt reactors or molten fuel salt reactors were the most difficult of all of these.

So we started a couple of efforts about a year, year and a half ago where we convened an IEA PIRT panel. In order to take the available information that we had at that time on these types of reactor systems, much of that coming from the MSRE that had been done at Oak Ridge National Lab and trying to characterize what would be those important processes for an over-power event or an under-cooling event and we don't have all the design details, but we said hey, we needed to get some basic information on what phenomena are going to be important so we can make a decision on what those tools were going to have to be able to do.

Now for those and for all of these and one way we call these pre-PIRTs because we don't have all that design information and we don't have all the accident scenarios available to us right now. We think we can understand what the most important phenomena are based on that information we have right now, but at some point we're going to have to revisit these PIRTs when those applications come in and we see do they have a DRACS system? What type of an RCCS system? Any other safety systems? Do they have a freeze lug or a drain tank and things like this that some designs may have and others won't have frankly.

DR. PETTI: So Steve, let me ask a question because you talked about salt, if you could. Was it focused on a thermal system? Or did it also include the past?

MR. BAJOREK: It includes past, it includes both, included both.

DR. PETTI: Because I think there are still technical issues that aren't necessarily accident safety. But I must say, I'm not convinced the fast salt system is controllable.

The delayed neutron fraction in a fast reactor smaller than a foot, I'm taking some of the delayed neutron precursors and moving them around.

Some of them are going into the gas space. Some of them are plating out on the tube. Then, I'm going to burn everything on this thing for 30 years. I've got all the uncertainties of cross sections.

How do you know what's going on?

MEMBER CORRADINI: And the fueling dynamic.

DR. PETTI: Right. I mean, that's nothing but accidents. Can I control them? You know, is that common?

MR. BAJOREK: Yes. You're going to like a couple of slides then towards the array where we start to take a look at that.

DR. PETTI: Okay.

(Laughter)

MR. BAJOREK: Anyway, these are the PIRTs that they can give away. I'm going to jump through a couple of these.

And if you'll allow me to, I'm not going to -- I've gotten a slide for each one of the gas-cooled, the sodium fast, the micro and the molten salt reactors.

Gas-cooled reactors as we've mentioned, we think we're in pretty good shape from NGNP. The design basis tools would be working at more like your loss of flow and your AOO types of events.

Whereas anything with chemical reaction or transport of the fission products, that would be handled more by MELCOR. I almost said NOTRUMP for some reason.

Okay, the --

MEMBER CORRADINI: The quiz is to who knows what that is in the meeting.

(Laughter)

MR. BAJOREK: Yeah. I don't know why, but. Okay. Sodium fast reactors, we're looking at scenarios that will involve loss of coolant with and without scram.

The unprotected loss of flow, or the EBR-II types of accident scenarios that were shown to be, shown to actually have quite a bit of margin in that system. Reactivity-induced types of transients.

Some of the things that are of interest and of concern, stratification. Okay. In the designs that we are looking at, we expect these to be more pool-type reactors as opposed to loop-type reactors.

Where things are in the vessel and that's where your natural circulation is going to occur when you have a transition from forced flow to natural circulation flow.

Stratification then threatens the vessel temperatures that may occur. Reactivity changes due to mechanical changes in the core structure, as that lower core plate expands or contracts, that controls your axial and your radial leakage from the core.

That's your biggest component of negative reactivity. So we need something that's going to be able to handle that.

Now, where things start to get a lot more interesting, I think, in terms of the newer phenomena, micro reactors, where you're concerned with, or essentially you have the reactor more or less steady state cooled by heat pipes.

Scenarios that have been identified in the PIRT for being of most interest, loss of heat sink, okay. You can also think of that as a loss of your heat pipes.

Okay. The heat pipes don't have any place to dump the energy. It's like the loss of heat sink. Inadvertent reactivity insertions. One that's sort of interesting for that is the possibility of a seismic event, where you may have a compaction of the core.

In a sodium fast reactor you're not so worried about that because the core is already impact, fairly compact. And you have wire wraps around the rods. It doesn't have a whole lot of movement there.

But some of the heat pipe designs, you've got a little bit more flexibility in the core. And if there's a seismic event or some other event that causes compaction, that could shrink that feature of negative reactivity from the axial and the radial leakage.

So, this relationship between the thermal-mechanical expansion, okay, and the leakage that you need to have for the neutronics starts to become much more important for this type of design.

MEMBER KIRCHNER: Are all of those designs fast?

MR. BAJOREK: I believe so, yes.

MEMBER KIRCHNER: Oh. So the one that I designed and built in the '80s was thermal. So, why are they going fast? Just to keep the size down?

MR. BAJOREK: Probably more compact energy generation.

MEMBER KIRCHNER: Yeah. I get it.

MR. BAJOREK: The ones that we've see have been fast.

MEMBER KIRCHNER: They're all fast. Okay.

MR. BAJOREK: One is --

MEMBER KIRCHNER: So all depending --

MR. BAJOREK: The exception is possibly in the thermal.

MEMBER KIRCHNER: All depending on leakage for reactivity feedback.

MR. BAJOREK: As you are wanting the same thing.

MEMBER KIRCHNER: As the safety means.

MEMBER CORRADINI: The public information from the recent released yesterday Strategic Capability Office implies that a lot of them are probably going to have to be thermal to get money.

MR. BAJOREK: Okay. These are the ones that have come to us primarily through the regul -- the RIS.

MEMBER CORRADINI: Oh, okay.

MR. BAJOREK: So the newer -- so, I know

that there's another call for proposals. And those types of designs, if they go thermal, that -- and we have to look at them, we will have to adjust.

But, I don't believe any of them have talked to us at this point.

MEMBER KIRCHNER: But let me back up here in the methods business, not in the reactor design business. And so let me see if I can make my comments that way.

From a method's standpoint, the capabilities you would put together and probably use, it doesn't matter whether it's fast or thermal.

MR. BAJOREK: If it were thermal, I would probably not care about the thermal mechanical expansion.

MEMBER KIRCHNER: Right. Right. Then it becomes a design detail.

MR. BAJOREK: Yeah.

MEMBER KIRCHNER: And an area of focus for the design, for the safety review, not -- but from a methodology standpoint, you would be --

MR. BAJOREK: It would make our job easier.

MEMBER KIRCHNER: Okay. But the methods that you envision for these micros would be -- it wouldn't matter whether --

MR. BAJOREK: It would be the same.

MEMBER KIRCHNER: Whether they're epithermal, thermal, or advanced.

MR. BAJOREK: Yeah. The differences would come in the neutronics methods. So, you wouldn't have that.

MEMBER CORRADINI: Let me ask Walt's question differently. If they're so small and so simple, wouldn't I want a simple analysis tool?

I wouldn't want to have an enormous mallet to beat something that is obviously calculably with a simple analysis, safe. Whatever that is.

MR. BAJOREK: I agree. And if, you know, in looking at these designs, and I think we have a figure going on that can show.

The main analytical question mark is going to be getting the energy out of what we call the monolith. Yeah, this is the structure that holds the rods, the control rods, the fuel, the heat pipes, whatever you have in there.

If you don't shut off the reactor for some reason, the drums don't rotate and you lose some of your safety systems, you're left with conducting that energy to the boundary, and either radiating it away or convecting it away.

That's probably the central issue for analysis for one of those. And by in large to being able to do conduction.

Okay, you know, there's always a question with 3D radiation convection. But, being able to do those things are well within our means.

MEMBER CORRADINI: Okay.

MEMBER REMPE: Before you leave, I have a question about the localized T-pipe failures. I was hoping I could say what I was interested in.

I think it is something like curd and axial offset anomaly and a steam generator, is what I'm asking about. And I didn't see anything in your document that looked like you were worried about any sort of deposition from this sodium that flows through these heat pipes.

Did the PIRT, the experts that participated in the PIRT say oh, that's not a problem? Or nobody brought it up?

MR. BAJOREK: I don't think it was brought up. I don't think it was a problem so much with crud. But if for some reason you were -- your heat pipe -- one heat pipe would corrode, fail.

You lose your sodium or your NAC or

whatever is in there, in that heat pipe. That region heats up, do you have a cascading failure that causes failure of surrounding heat pipes.

MEMBER REMPE: Okay. And I --

MR. BAJOREK: That was a concern. Now whether that failure is due to crud, corrosion, or any other mechanism, I don't recall whether they tried to blame it on that or anything.

MEMBER REMPE: Okay. So, somehow or other you'll identify data that's needed. Not whether it was sodium flowing through a pipe, and whether it fails because some stuff deposits or it corrodes will be addressed as your data needs somehow or other.

MR. BAJOREK: Likely that or a performance data of the system. And probably it depends on whether, you know, how much margin you have after you fail that heat pipe. Okay?

MEMBER REMPE: Thank you.

CHAIRMAN BLEY: Steve?

MR. BAJOREK: Yes?

CHAIRMAN BLEY: Just kind of going back to where we were in the beginning. But, you have on each reactor type, you have a section that identifies the information gaps, knowledge gaps.

MR. BAJOREK: Um-hum.

CHAIRMAN BLEY: And you have the evaluation using your PCMM, which is the maturity level. The thing, and a few of us were talking about this before we got here.

The thing I don't see spelled out anywhere is how important to a safety conclusion filling the knowledge gaps, technical gaps, would be. And how important to the same end improving the maturity would be.

And likewise how much effort it would take to get rid of a gap or improve the maturity?

MR. BAJOREK: I understand. And we discussed that before. And I know some of those rank --

CHAIRMAN BLEY: But you, I mean, you must have ideas on that.

MR. BAJOREK: Some of these rankings we bias low.

CHAIRMAN BLEY: Okay.

MR. BAJOREK: From this point. Because, you know, we -- especially when we're not sure of some of the aspects.

I think one of the things that it's going to be very important as we go along, is going to be identifying how much margin there is. The more margin

--

CHAIRMAN BLEY: That's the only way you answer these questions.

MR. BAJOREK: If you have lots of margin, it affords you larger uncertainties in some of the phenomena. In PCMM, down at the very bottom is uncertainty. Okay.

CHAIRMAN BLEY: Yeah.

MR. BAJOREK: In the sensitivity analysis.

CHAIRMAN BLEY: And very few do well there.

MR. BAJOREK: And it's a tough area. We do more of that in light water reactor because that's -- that has been built into some of the best estimate things.

But, it's not quite there for the light water reactors. Now, what we might be able to do, is rather than using a full-fledged statistical method for uncertainties, okay, you may be able to, if you've identified the phenomena, which are questionable, okay, and you bound those, and you do a few sensitivities, you don't have to go there.

CHAIRMAN BLEY: Yeah. That's kind of where I'm hanging.

MR. BAJOREK: And then this --

CHAIRMAN BLEY: What my worry is, we put a whole lot of effort into getting rid of these gaps. And to im -- maybe improving the maturity in places where we don't really need.

Well, it's the same thing you guys were talking about earlier.

MEMBER REMPE: Um-hum.

MR. BAJOREK: Yeah. And this same --

CHAIRMAN BLEY: And if we don't do that first, we're going to spend all that money doing it. And then say, oh, we didn't have to do all that.

MR. BAJOREK: And likewise, the second element of PCMM is the physical models and phenomena.

CHAIRMAN BLEY: Yeah.

MR. BAJOREK: Do I have to be state of the art? You know, that highest rank? I think the staff will probably say no.

Because there again, if I have a process that I can reliably bound, okay, and be conservative with it, we're done. Okay. We don't have to do it.

Now maybe the applicant may not like that decision. Okay. Or they may not like that loss in margin, but then that's for all of us, the entire industry to chase in the future.

But it should not hold up the licensing

effort.

MEMBER MARCH-LEUBA: Yeah, but in this bus -- there are two customers in this business. One is the safety of the public.

MR. BAJOREK: Um-hum.

MEMBER MARCH-LEUBA: Which is what you care about. Another thing is the economy.

MR. BAJOREK: Right.

MEMBER MARCH-LEUBA: So, if bound in there, and what I call cheating by bounding of the material, I don't have to do the notice, because I know it's all 0.2. I satisfy the safety to the public, you stop.

They want to talk the measure design, let them do the more sophisticated one.

MR. BAJOREK: So that's identifying the gaps. Fulfilling those gaps, okay. And increasing the PCMM until we're satisfied with it, does not mean we have to go all the way and make everything perfect.

CHAIRMAN BLEY: That's right.

MR. BAJOREK: It's going to happen --

CHAIRMAN BLEY: But even if they push, if you can do a simpler approach, that's all you need to do.

MR. BAJOREK: That's right.

MEMBER MARCH-LEUBA: Now in that sense in the PCMM, I would add an extra role, which is bounding themselves to it. To be really specific about it.

MR. BAJOREK: Finally the MSRs, and I think you're getting to a couple of the comments that come up. Yeah, these are much more difficult for us to deal with.

And why it's on our plans right now, is because we see this more further long range. It's going to take longer for us to get ready for it.

MEMBER CORRADINI: But the left hand side is not the same as the right hand side. If I've got basically solid fuel where I know where it's supposed to be, it's a whole lot different than if it's circulating around on the right.

MR. BAJOREK: But many of those things on the left also apply over to the right.

MEMBER CORRADINI: Yeah.

MR. BAJOREK: So you have the right hand side, which this is what came out of the PIRT. So, I try to be very religious as to what those authors did on this.

But, if you start looking at what's going on over in the right hand side, the key on that is the delayed neutron precursor motion. Of course that's --

MEMBER CORRADINI: Steve, I want to make sure of the program you keep on talking about. Is this the BNL thing with Dave Diamond as the lead?

MEMBER REMPE: The precur --

MR. BAJOREK: Yes.

MEMBER CORRADINI: BNL-114869?

MR. BAJOREK: There's two of them. Yes, Dave Diamond over on the right hand side. And over on the left side was Farzad Rahnema from --

MEMBER CORRADINI: Oh. I guess I'm not familiar with that.

MR. BAJOREK: Georgia Tech. Yes.

DR. PETTI: It was an FHR. It was an FHR.

MEMBER CORRADINI: Oh, the IRP?

MR. BAJOREK: That was part of the IRP. Yes.

MEMBER CORRADINI: Oh.

MR. BAJOREK: These came from two separate efforts. So that's why they're not exactly saying the same thing.

But, I think if you look at what was summarized in one, you'll see many of those same things over in the fuel side.

MEMBER CORRADINI: Okay. Thank you very much.

CHAIRMAN BLEY: We ought to see, I think, everything on the fuel side that shows up on the coolant supplement side. And I think you do, only under different names.

MR. BAJOREK: Yeah. I agree. I mean, but this is what came out of the PIRTs and I tried to stay with it.

But yeah, the fuel salt, in getting to your question, is difficult because of the delayed neutron precursors. If it happens you have a loss of force flow, now your precursor is being released elsewhere.

Now they may be back in your core as that, and not out near the heat exchange. Okay. That could be a sudden insert of reactivity.

So that's one that we're looking at. Salt chemical composition. Okay. These are often eutectics of two or three different salts.

They behave as a single fluid as long as you stay with the eutectic composition. You start dumping in fission products or something else, your properties can become very nonlinear.

Both PIRTs identified physical, thermal physical properties as being a major concern. Because in some cases you don't know that thermal conductivity of viscosity too plus or minus 30 percent.

For water we're kind of used to .03 percent or something very good. And that's not to say anything about things like melting temperature or freezing temperatures.

Even those have quite a bit of variability from the data.

DR. PETTI: So, I have a couple of questions. Just in the layout of the three volumes. In the fuel chapter that we haven't seen yet, will fuel salt be in that fuel chapter?

Where is -- where are they going to capture performance of the fuel outside of severe accident space?

MR. BAJOREK: The staff is working on fuel qualification for salts. That's early in time right now.

I don't know if that's going to necessarily show up in volume two if we want to get this out in the next few months.

DR. PETTI: The other thing --

CHAIRMAN BLEY: There ought to be a place holder

DR. PETTI: There ought to be a place to make sure. But the other thing that doesn't come through here, the effective impurities in the salt, really, really important.

You form slag if you -- there's old experiments clogging pipes, little pipes, because they did stuff really small.

But, it can change the chemistry. There's also stuff that can happen, so there would probably have to be tech specs on impurities, you know, those things.

MR. BAJOREK: I'm going to jump ahead.

DR. PETTI: Okay.

MR. BAJOREK: You know, yeah, we recognize that this whole chemical composition represents a very major gap for the fuel salts.

And you know, what I'm trying to show here is essentially we've got our salt, which is we're depleting the fissile material. We're building the fission products.

Some of those are gaseous and they go off into a cover gas. Why your tritium is generated there, and it goes off into the cover gas. And that actually maybe your largest source of fission products, one of these.

As you go through whatever that system is, whether it's a loop or not, now you have to deal with chemical reaction as these fission products decaying and possibly become chemically reactive, some of them will go out in your filter system.

Some will, you know, some will play out and become sediment in the ones which are insoluble. And these fuel cycle -- the ones that have fuel cycles, want to have some type of an operation where they're adding fissile material.

Perhaps not even the same fissile material that you started with, but something else. And bleeding off some of the fission products.

So, you know, we recognize this is a very, very important gap. I called it inventory control gap because it's also very important for non-proliferation.

Because it's a liquid fuel, you need to know the fissile material content and where it's at to a much greater accuracy then you do for the solid types of fuels.

DR. PETTI: I have another question. Do you guys care about the carcinogenic nature of the beryllium and the beryllium salts?

MEMBER CORRADINI: That's not radioactive.

DR. PETTI: I know it's not radioactive. But it's carcinogenic for worker safety. Is that, I mean, that's the -- and I don't know if some of the metrics are.

Because I haven't look at those. They are nasty and I don't think so.

MR. BAJOREK: In terms of the experiments at the university, beryllium is an ugly actor.

DR. PETTI: Right. So, not just theory.

MR. BAJOREK: And you know, there's plenty of experience with those.

(Simultaneous speaking)

MEMBER CORRADINI: And then my only point is in terms of doing experiments with salt on campus, the presence of beryllium chains is a whole different set of procedures.

DR. PETTI: And if you go back and read the MSREs, the literature, the beryllium flow is a preferentially deposit in cold spots. Valve stems, all the places where you need to get to do maintenance.

And it stays RIC. And so it will just go airborne, and you know, the toxicity limits are quite low for beryllium.

MR. BAJOREK: I think what's gotten more of our attention has been the tritium production. Any time you have a lithium-bearing salt, you're going to give off lots and lots of tritium.

And that's --

DR. PETTI: All over the rains, and you will get sick, so.

MR. BAJOREK: Yeah. You have to --

DR. PETTI: You still get a fair amount of tritium however.

MR. BAJOREK: Yeah. Okay.

MEMBER REMPE: So for the MSRs are you going to do chemical effects also in your CRAB package?

MR. BAJOREK: No. We are going to talk with the Department of Energy, and let them know that this is a -- this is an issue that the industry has to solve.

CHAIRMAN BLEY: Yeah.

MR. BAJOREK: Not just us. So we want to work with the Department of Energy to come up with the right tool in order to access what that chemical composition is in a molten fuel salt reactor.

We see that as something that the industry needs to bring to us. Not us to be the only sole developer.

MEMBER KIRCHNER: Steve, when I looked at your overall rankings from your PIRTs and such, I mean, intuitively they made sense. But, what I was missing, how should I put it, degree of difficulty.

Readiness is one thing. Maturity is one thing. But degree of difficulty is another aspect. And my goodness, we just were covering a few of the issues that you would have with a molten, with a circulating fuel system, whether it's a salt or whatever it is.

Does that -- it doesn't quite come through. And I know your job is not to rank the designs. But -- methodology.

But also the methodology could be a difficulty. Is a --

MR. BAJOREK: We don't --

MEMBER KIRCHNER: Is a much bigger reach I should -- it's not impossible. But the reach and the resources to get there are going to be significant.

MR. BAJOREK: In Volume One for each of these gaps, we've identified a task. Now we don't put it in this report because it's a public document.

But elsewhere we've identified what it would take to resolve that gap in terms of manpower and resources. So we can get an estimate on those numbers.

Although in some cases, it's difficult to get a very accurate sense of that. But we do have sense in what that is now.

MEMBER KIRCHNER: Do you -- this is also more in the review of the actual concepts then the methodologies. But, they are related.

Do you have a sense that the people realize the degree of difficulty that they're taking on?

I mean, a realistic sense of what is needed? Because the bar is going to be very high. And convincing -- and I'm just talking about public safety.

I'm not talking about economic, you know, we're talking about things, chemical effects and such are the nightmare of maintaining an operating system and such. So those are issues that they're going to have to deal with.

But just the bar in the regulatory arena, how high that is. Is there a sense there of how -- I mean, there's a lot of public pronouncements about how safe these reactors are.

But, that remains to be demonstrated in many cases. The reach -- I'm trying to just say that yes, the PIRTs make sense. And your ranking makes intuitive sense.

But missing was the degree of difficulty, the reach for that -- for some of the methodologies. And I'll try and confine my comments to methods. It's substantial.

MR. BAJOREK: I agree. And I, you know, and I can't -- I don't want to comment on any of the applicants.

But, I think you see a range of those that really have an understanding of what they're getting into. And some that have a very good academic idea, but don't quite understand the licensing process and the processing hurdles that they're going to face.

MEMBER KIRCHNER: Okay.

MEMBER REMPE: So, on your report, even with this public version, I saw tasks identified with little numbers and alpha-numeric type thing. And are those tasks that NRC intends to complete?

Or something that you've identified and you hope DOE coughs up the money for? Or what do I -- what are those tasks?

MR. BAJOREK: It's a mix. We're working with the Department of Energy to resolve what those are.

We see a lot of the developmental

assessment that goes into the code as part of that overall development process. And we're going to work with the Department of Energy to complete that.

There's some validation that we're going to keep for ourselves. That's training. That's how we educate ourselves on how these tools work and how these systems are going to behave.

Things which are more licensing oriented like developing the reference plant models, that probably falls on us. So, it's a little bit of a mix.

MEMBER REMPE: Okay. And so earlier you talked about that you have the source coding. And you can make changes to it.

Are you going to validate your version? Or will -- how does that, I mean, if you start making changes to the Code, and then it's not the same Code as what the DOE folks have right?

MR. BAJOREK: Eventually you need to do your validation with a strictly frozen code. I mean, that's the language in some of the guidance.

Although no one ever does that. You get as close as you possibly can. But yeah, if we're going to make a major change, we need to rationalize that hey, it doesn't change things.

It fits with the other parts of the code where we'd have to redo some of that assessment.

MEMBER MARCH-LEUBA: Besides the time to get it done, I mean, you can do it in C++ or whatever. You can kind of keep it to that.

If you decided that you needed a change, because you will not get an accurate result, how come you accept theirs? And it becomes an issue of philosophy.

MR. BAJOREK: Well, I'll defer that one until we do some more of the validation. If we see something that is a clear flaw that we need to change the model, we may go back and say hey, you need to put this model in or change this model, or this needs to be fixed.

As you go through any validation, you're going to find some flaws. If you don't, you haven't looked at it right.

MEMBER MARCH-LEUBA: And I would like -- you and me are cool guys, we like that. But I would put a very high bar into making modifications to the Code.

Unless it's absolutely needed. And if it's needed, then you do.

MR. BAJOREK: Understood. Okay. As we go through the PIRTs, we can summarize some of the major differences between what we have in light water reactors and non-light reactors in several overall categories.

The need to look at stratification. This idea of thermal-mechanical expansion, especially in the fast reactors.

The idea of a large neutron free path that you see in some of the graphite reactors and in the fact reactor, with the control rod over here, can affect my fuel over here, and my power is going to reach.

Okay. We don't have that in a light water reactor where you have control rods in almost every one of those assemblies. Transport of the neutron precursors, solidification and plate-out in the MSRs.

And when we have a monolith-type structure, whether it's a micro reactor or a large graphite series of blocks, this idea of being able to evaluate the conduction of energy from that block to the boundary where it's either going to be radiated or convected away.

These are things that if we look at our NRC legacy tools, we either cannot do those, or adding those features would be very, very difficult, if we could do those at all.

So that's why we have looked elsewhere at the DOE, the NEAMS, the CASL, Argonne tools. We've scanned through all these.

We've contrasted these with our PIRTs and are recommending a small package of codes that can be used for each of these reactor designs. Now, you don't use all of these codes at the same time.

MEMBER CORRADINI: Good.

MR. BAJOREK: The light water world is sitting over here. Gas cooled reactors are over here. Liquid metal and molten salt is all down in here.

Now they're held together by the MOOSE and the MOOSE framework. The MOOSE by itself is a computational tool. It allows you to do thermal-mechanical expansions among other things.

And conduction in complex geometries. It also has the solvers, and the data transfers, and all those things that the software engineers really get a kick out of kind of doing that makes all these other tools work.

So we're using it for the thermal-mechanical expansion. But we're also using it based on its ability to couple with other tools.

We've already completed the work to couple TRACE into the MOOSE. So if we have a gas cooled reactor, and it has a ranking cycle in there, okay, we can use the DOE tools to do the primary system.

And then transfer that information over to TRACE when we have to do a ranking cycle for the secondary, tertiary systems, or things like decay heat removal systems, or reactor cavity cooling systems that operate with water.

Other systems might be easier for us to try to do the modeling with TRACE then to use these tools. But the idea here is to use a limited set of the Department of Energy tools for each of these types of reactors.

And give us the flexibi --

MEMBER MARCH-LEUBA: Wait, wait. No more. One thing I see missing here, which is missing in our light water reactors too, is the steady state core simulator. The coolant to simulate.

Most of the things you're describing, especially the more mechanical expansion of flexible activity, don't need to be handled on transient basis. It's a lot easier to do step wise steady state.

And you can have a lot more confidence in the results. And I see it, I mean, certainly you wait too long for code, do we have a phase that caution the different area. Because we definitely need it.

MR. BAJOREK: Okay.

MEMBER MARCH-LEUBA: And I think many of the difficult problems that you're describing can be solved, because all this reactor is going to have, or some of them are going to have a long time constant?

MEMBER KIRCHNER: Very long, yeah.

MR. BAJOREK: Yeah.

MEMBER MARCH-LEUBA: And you can do quasi-steady state. And it's very easy to do expansion with steady state. Where it's very difficult to do on a platform.

MEMBER KIRCHNER: And some will have very short time constants.

MEMBER MARCH-LEUBA: Yeah.

MEMBER KIRCHNER: And those could be dominant. Or of great importance. But, I'm going to reinforce your comment and say that you could still -- let's choose one concept where this is going to be really a critical factor in your review.

And that's the fast, the sodium fast reactors. They're not going to build something as small as the EBR-II. It's not commercially viable.

Well, maybe some micro design might work in Alaska or something. But, notwithstanding that, so they are going to push the envelope.

And they are dependent on leakage for the reactivity feedback. And it seems to me that to do that dynamically is extremely difficult in a systems manner.

But as Jose is suggesting, if you had really good physics package, you could step your way through this. And bound whatever they present to you without having a full dynamic capability of thermal-mechanical interaction and neutron.

MR. BAJOREK: Yes.

MEMBER KIRCHNER: So, I'm reinforcing your point. But it maybe for a very short --

MEMBER MARCH-LEUBA: Short term costing.

MEMBER KIRCHNER: Critical feedback affects --

MEMBER MARCH-LEUBA: Everybody was affecting.

MEMBER KIRCHNER: In the fast spectrum system as an example.

MEMBER MARCH-LEUBA: Well, I was saying before that you can bound and approximate the results and have confidence in the new calculation.

Something that is very dear to my heart and my colleagues will hear more this week about this, shut down margin. And if you -- you might have shut down margin for new reactors.

You have to demonstrate that beyond a shadow of any possible doubt. And if you do it with a transient with expansions and any kind of reactions and all, you cannot do it.

MR. BAJOREK: We would want to try to keep these models as simple as we possibly have, and it comes into --

MEMBER CORRADINI: So can we go back to the cartoon? So let me pick on one, so let's pick on Nek5000. What I heard you say was that you were going to use these tools in selected areas where you need a key piece of information that the historic NRC evaluation models can't give you.

MR. BAJOREK: Right.

MEMBER CORRADINI: So give me an example of what Nek5000 can do that you can't -- that you need to do, that important to do that you can't do with the current part of the evaluation models as an example.

MR. BAJOREK: Well, let me rephrase and let me characterize the CFD that's on here. Even though Nek is part of the loose framework, okay, our anticipation is we would only be using CFD to help us benchmark other place, other codes where we have question marks in looking at a region, a stratification, a plume, things like that.

We would not use those in a coupled transient matter, okay. They might be a side alone calculation where we take some information to look at details of something, but we would not want to get into the business of having to run a coupled calculation with Nek, or FLUENT for that matter.

MEMBER CORRADINI: But can you give me an idea of a piece of physics either in the gas world, the sodium world, or the solid fuel mold cell world, I'm not going into liquid fuel, that I need Nek5000 to answer a question that's crucial?

DR. PETTI: I can tell you one if Steve can't.

MEMBER CORRADINI: No, but he's on --

DR. PETTI: I know.

MR. BAJOREK: You're not the guy. I'm the one being quizzed. Well, actually I would have gone to the mold fuel salt reactors first because you see a lot of pool. You know, some of those have a pool configuration and you need to understand what that recirculation is going to be.

MEMBER CORRADINI: Okay.

MR. BAJOREK: The other place is going to be a stratification that you get in a sodium fast reactor.

MEMBER CORRADINI: So you're worried about thermal striping issues and fatigue.

MR. BAJOREK: Right.

MEMBER CORRADINI: Okay, that's the only one that pops in my head.

MR. BAJOREK: We'll show you in just a second why I don't think we're going to need CFD coupled in because we have some of that capability already available in one of the other tools.

MEMBER CORRADINI: Okay.

MR. BAJOREK: So we can do that.

MEMBER CORRADINI: Okay.

MEMBER REMPE: Before you leave the cartoon, I have a different question, but did you want to say something first?

DR. PETTI: I was just going to say in a gas reactor, a pressurized conduction cool down can lead to a depressurized conduction cool down if the natural circulation flow will fail the control rod guide tubes if more hot helium goes up than you think, and that's --

PARTICIPANT: Okay, so normally the flow is the --

DR. PETTI: Right, but it's a

multidimensional problem. You might want to do CFD when the vendor comes in and says, "Yeah, we've looked at it." It's a pretty complicated problem. You might want to do a side calculation to convince yourself from an action and profession standpoint, but, you know.

MEMBER REMPE: But then you can just put a hole in MELCORE and look at those --

(Simultaneous speaking.)

MEMBER REMPE: But anyhow, on this, for validation, which I know is on the next slide, but when we talked to the DOE code proponents in prior meetings, they acknowledge that the governing equations in BISON and FAST differ, and to validate the model, you might want to get different data for the BISON code versus the FAST code, and is that considered in your evaluations because you --

MR. BAJOREK: That will be in volume two.

MEMBER REMPE: Okay, so you'll --

MR. BAJOREK: Yes.

MEMBER REMPE: -- carefully explain to me why you really need BISON and you can't just get away with FAST?

MR. BAJOREK: We will carefully explain it.

MEMBER REMPE: Knowing how slow I am, okay, thank you.

MR. BAJOREK: Okay.

MEMBER CORRADINI: So this is a nice cartoon to ask that question about the one thing I don't understand, which is reactor physics. Tell me about MAMMOTH relative to SCALE or maybe I don't understand the question.

PARTICIPANT: PARCS.

MEMBER CORRADINI: Well, I was going to go to -- PARCS is probably what I want. That's the nodal code, right? So if I understand this correctly, SCALE provides the macroscopic cross section that could be used in PARCS or could be used in MAMMOTH?

MR. BAJOREK: That's a -- there would have to be a transfer developed --

MEMBER CORRADINI: Okay.

MR. BAJOREK: -- for that, but --

MEMBER CORRADINI: So what does MAMMOTH provide that you can't do in PARCS? What's the gap in PARCS?

MR. BAJOREK: We're going to come up.

MEMBER CORRADINI: Okay.

MR. BAJOREK: We have a couple of slides coming up on that.

MEMBER MARCH-LEUBA: Before you move out of the cartoon, can you tell us what's the state of development? If you wanted to use today, TRACE, I know you can't use it. How about the others?

MR. BAJOREK: I would venture that SAM, as far as we're along the PRONGHORN, and it's used right now.

MEMBER MARCH-LEUBA: Everyone talks very well about BISON, but can you use it now?

MR. BAJOREK: We could use it. I mean, for the design basis part of this, we're mainly looking at the heat release. We need the stored energy, the thermal conductivity. There, they would be kind of comparable.

MEMBER MARCH-LEUBA: But is --

MR. BAJOREK: For looking at fission gas release and other things that a fuel performance individual wants, I'm not sure, you know, who's got the lead on that right now. That's not something I really need for the design basis model at this point because --

MEMBER MARCH-LEUBA: Well, how many years will it take you to validate those codes so you can use them legally?

MR. BAJOREK: I think everything we do is legal, but --

MEMBER MARCH-LEUBA: Because we might get one of these reactors anytime soon, right? I mean, if you have a 10-year validation plan, that won't work.

MR. KELLY: Since Steve did not mention MAMMOTH, I will. It is currently being used to design experiments in the TREAT reactor. That's one of its main reasons for being.

MEMBER MARCH-LEUBA: What is the advantage versus PARCS?

MR. KELLY: It goes far beyond the capabilities in PARCS. It goes full transport, SN transport, all the way to point connects with lots of different flavors in between. I'll show a couple of little examples when I get to my part of this.

MEMBER CORRADINI: And that's necessary for certain steady state conditions or setup for certain transients?

MR. BAJOREK: This is where some of the simplification comes into play.

MEMBER CORRADINI: Okay.

MR. BAJOREK: In some cases, you're going to need some of that detail, but you don't want to expand it everywhere, and one of the features of MAMMOTH is having the, being able to change your nodalization, to put the details where you want it, okay, and use diffusion in other places, okay.

It's taking it a step beyond PARCS and giving you that accuracy, but without the sacrifice of having to nodalize the heck out of this thing all over the place.

MEMBER REMPE: And when you came up with that justification, there's some safety issues that it will help you address you might not see otherwise?

MR. BAJOREK: Yes.

MEMBER REMPE: Such as?

MR. BAJOREK: Such as getting your power distribution correct. Otherwise, you would not be getting the correct peaking factors and power across the core. You'd probably be under predicting it in the fuel and over predicting the behavior of the control rods.

MEMBER MARCH-LEUBA: But you're talking two percent or 150 percent?

MR. BAJOREK: More than several hundred PPM, PCM, excuse me.

MEMBER MARCH-LEUBA: PCM.

MR. KELLY: Yeah, and the other main point which is being overlooked is it's a MOOSE-based code. We haven't really brought to you what MOOSE is, and what MOOSE-based codes are, and why they work so well together.

If you look at any one of these codes, whether it's PRONGHORN, MAMMOTH, SAM, or BISON, probably 90 percent of the actual lines of code they use are in a MOOSE framework. They only have that small few percent of constitutive models or maybe a slightly different conservation equation for that particular application.

That's what's in that app. Everything else is this large, if you will, library called MOOSE. It's what enables all of those codes. None of them stand alone.

MEMBER MARCH-LEUBA: You're scaring me now. I mean, how are you going to validate a license then if you have so many meaningless lines of code?

MR. BAJOREK: Just to get to the next slide, we've already done and seen some of the validations giving us enough confidence that, yes, you can take these packages and complete the validation. They're getting us the physics that we need.

We think it has the flexibility to simplify the modeling of the system so we don't have to have inordinate detail, and we can then run those on computers that don't need high performance computing systems.

We don't want to have to run routinely on 10,000 core HPCs. We want them to run on the things which are available to us and, you know, the wonderful equipment we have to work here, the staff, so, yes, we want to keep that low.

And the calculations we've done so far, several of us are running these codes here. We're running them on these boxes right now and it's given us the confidence that, yes, we can run these on our platforms. They're flexible enough that we can look at lots of different scenarios and situations.

There is some V&V out there. As I say, it's critical and it represents a gap, okay. You know, we're trying to identify the gaps. We can't solve them today, but we're going to identify those and we'll lay out a plan to go get those the rest of the way.

MEMBER REMPE: So with your power peaking factor example, in the real industry, they do a lot of uncertainty evaluations to try and make sure they have sufficient margin because the agency requires them to do that. So you think you're going to get such validated codes that you won't need that?

I mean, in your little evaluation, PCCM, uncertainty was an area where it said, "Well, we're going to think about that later," and I'm just kind of wondering about --

There's still going to be uncertainties and how are you going to address that, and is this additional confidence in power peaking factor so important when you actually do uncertainties or require an applicant to do uncertainty evaluations?

MR. BAJOREK: I'm kind of thinking back to light water days and, you know, I think our fuel division always used to keep eight percent margin in their pocket to cover their uncertainties. I would expect the designers to continue to do something like that right now.

What you need to do for metallic fuel or for the TRISO fuel, I really don't know at this point. However, our goal is to develop the tools so if an applicant comes in and they say, "Hey, this is the power distribution," do we agree with that or do we think they're off by a large amount?

MEMBER MARCH-LEUBA: If I could ask you more, the staff concerns. We know when we went to MELLLA+, everybody went to one particular guy in research.

MR. BAJOREK: They still do.

MEMBER MARCH-LEUBA: I'm just giving you just an example. How many staff members do you have that are familiar with these codes?

MR. BAJOREK: Right now, we've got a group of five or six of us that have been dabbling in it. Joe is certainly our point on this.

MEMBER MARCH-LEUBA: So you have a critical mass?

MR. BAJOREK: We have enough that we're, within the next month, we're going to have so many running on each one of these codes.

MEMBER MARCH-LEUBA: So you have more than one guy to do that and carry all of that with you, so that's another consideration.

MR. BAJOREK: You know, we're running them. Joe runs them. I run them. We have a couple of other guys running these as well.

MEMBER MARCH-LEUBA: You need to expand that.

MS. WEBBER: Yeah, but if I could say, I mean, that's a big concern that we have because it's a very, I would call it small industry with the core capabilities of code development, nuclear engineering, safety review, and so that's something that we're going to try to address it at a higher level.

MEMBER MARCH-LEUBA: I would to throw a different concept in there. Have you considered talking to DOE and segregating a group of already qualified individuals that are not going to have a conflict of interest and they can do it for you?

MS. WEBBER: Yeah, we've been also talking about this potential for conflict for interest, and so it's something that we obviously need to assess.

MEMBER MARCH-LEUBA: But they will segregate a number of experts for you so they won't have the COI when it comes time to do it.

MR. BAJOREK: Our goal right now is to start small, get people in the staff familiar with some of these tools, get help from the labs, okay. We're not going to use these as a black box, and augment the staff and our understanding depending on how the licensing and the market shakes out, okay. We're trying to get ready for everybody right now. That may not be the case in a couple of years.

MEMBER MARCH-LEUBA: It's more cost efficient to use experts that you have to pay that are not being used and are already trained, and DOE covers 50 percent of the cost anyway, so, but if you don't plan ahead and you don't consider conflict of interest.

MR. BAJOREK: Okay, we've outlined a number of things which are different about the non-LWRs. We've described these as things which would be difficult, if not impossible to build into the NRC legacy tools. What Joe is going to do in a few slides is just kind of give you some examples of the types of things that we're trying to take advantage of in these tools. Joe, you're way behind schedule, so.

MR. KELLY: As usual, and the first thing I'll do is go back a slide once I figure out how the mouse works.

MR. BAJOREK: Yeah.

MR. KELLY: So I have five examples I'm going to give, and my job in all of this has been to start learning how to use these codes and then start bringing some of that knowledge back to the NRC and try to help mentor other staff in their learning, and that process is just beginning.

Like Mike, I knew absolutely nothing about reactor physics a short while ago. Some may still question whether I know anything or not, but I'm actually doing reactor physics calculations for the first time in my career, which speaks somewhat to some of these tools.

But the five examples I'll give are

multiphysics coupling, geometric fidelity, advanced equivalence methods. I'm going to talk about a multi-scheme capability and also a 3D reduced order flow model.

And the idea here is we want to be able to model the salient features of these designs without having to do some kind of simplifications whose impacts we can't quantify.

So I want to be able to actually model what's there, and then we can get from that like an intermediate resolution solution, so not any high resolution, high fidelity.

We're talking more intermediate resolution in a model that only requires modest computational resources, things like Steve is always quoting a 32-core desktop, that kind of machine, grabbing a couple cords off of the cloud server that the NRC has access to.

CHAIR BLEY: Joe, let me interrupt you for a second first with a comment. You just scared me a bit saying you knew nothing about this and now these codes make you think you do.

(Laughter.)

MEMBER CORRADINI: We didn't say that.

CHAIR BLEY: We've chewed up a lot of the time asking all of you questions, and we've got about 10 minutes to go. We can take longer than that, but I'm wondering the best way to get to the end of this presentation and get to the other one.

Maybe we don't need to go through all of the examples. You know, pick one or two that you think illustrate what you wanted to illustrate with the examples and then go onto the --

MR. KELLY: This is actually -- that's perfect.

CHAIR BLEY: Okay.

MR. KELLY: And I'll do so, and this is a good one to start with. The letter M in MOOSE stands for multiphysics, and this is an example of multiphysics coupling.

Over on the left-hand side, you see a schematic from the small SFR meshed with SAM, which is a system level thermo-fluids code. We're modeling the core here with five channels. Over on the right is a cross section of the core which we've modeled with MAMMOTH.

MAMMOTH would be the controlling app and would calculate the 3D power distribution and feed it back to SAM. SAM would then feed back the temperatures and densities. Because they're both

MOOSE-based, they use the same transfer functions that are part of the MOOSE framework. That makes that easier.

Now, at the bottom, and this is something particular that's already been discussed for SFR, is one of the primary negative feedbacks is the radio expansion of the diagrid due to thermal effects. In traditional SFR analysis, you don't normally do that calculation.

What you do is you link the reactivity feedback simply to the core inlet temperature. Well, in something like the pipe micro reactor, there is no TMO. What you have to do is solve the coupled 3D and coupled thermal expansion problem in order to see how big your core has grown to see whether your axial or radial leakage effects can turn the reactivity down.

In MOOSE, there is a tensile mechanics module where it's just part of the built-in framework, and so we have used this in a coupled simulation with SAM to get the radial expansion of the diagrid during an unpredicted loss of flow. Then we fed it back in that calculation we've done so far, which is point kinetics.

MEMBER CORRADINI: So maybe this is not the right question to ask at this point, but when would you stop using this and move onto something different because this is, we were identifying this as design basis? I'm still struggling here. I would do this until I got to temperatures for reactivity insertions that would take me into what would be fuel damage?

MR. BAJOREK: Yes, or fail the vessel.

MEMBER CORRADINI: Okay, so at that point, I would defer to a different calculational approach, calculational tool set, and allow for more bounding, I use the word bounding, but a little more conservative calculations?

MR. BAJOREK: Yes.

MEMBER CORRADINI: So is this --

MR. KELLY: And still the thermal part of the transient you could extend beyond to use as a benchmarking tool as opposed to something Joy said earlier.

You can use an intermediate resolution tool to benchmark the thermal and mechanical effects that you would see and a lower resolution MELCORE time, but everything having to do with fission product transport and release --

MEMBER CORRADINI: Would not be here?

MR. KELLY: Right.

MEMBER CORRADINI: Okay.

MR. KELLY: Not be.

MEMBER CORRADINI: So the change, the switch over point would be I'm approaching reactivity changes or I'll just say temperature or strain limits on the fuel such that I'm expecting failures, and therefore I'm not going to do it here. I'm going to have to do it in a different tool set?

What I guess I'm -- what I'm trying to say, and I'm just thinking it through, what I'm saying to myself is I would use this tool set potentially for design. I would use this for, in the world of light water reactors, AOOs and mild transients, but as soon as I progressed out of that, this tool set is not the one to use to go beyond that?

MR. BAJOREK: An example might be for gas-cooled reactors. We sort of think that limit when we start to pop the TRISO is about 1,600 degrees C. This tool would go through and show us that hopefully you do have margin to 1,600 degrees C.

However, if we did an accident scenario with multiple failures that drove you through and the design basis goes, "Hey, your whole core is up at 2,000 degrees C," that's an accident scenario, then we would turn over to MELCORE.

Do you repeat that because now you're going to have, make a conservative estimate on the fission product release and transport out of confinement?

MR. KELLY: It's not so much, you know, T equals so many seconds.

MEMBER CORRADINI: Yeah, yeah, yeah.

MR. KELLY: We stop and go. It's that MELCORE would then redo the calculation. It would be an event selection process then.

MEMBER CORRADINI: Okay, fine.

MR. KELLY: -- identifying where MELCORE needs to go.

MEMBER MARCH-LEUBA: But if you reach these failures on a DBE, what you should do is return, reassign to sender.

(Simultaneous speaking.)

MR. KELLY: Remember Steve changed the definition here. He didn't say DBEs anymore. He said DBE codes, and a DBE level code will be used for multiple failures that are way out and beyond DBA specs.

MEMBER MARCH-LEUBA: I'm thinking --

MR. KELLY: But it's because of the margin they don't become major releases, so we can handle them with --

MEMBER MARCH-LEUBA: I'm thinking how long it took to get some MELLLA+ calculations on a light water reactor with established codes that we understood they were. Can you really do these calculations in an 18-month review cycle?

MR. KELLY: Well, there's one big difference. All of the codes you're talking about are serial in nature. They run on one core. These codes are highly parallelizable. If I run something like MAMMOTH --

MEMBER MARCH-LEUBA: No, no, I'm not talking about CPU time. I'm talking about the number of iterations it took to get it to merge.

MR. KELLY: Oh, okay.

(Simultaneous speaking.)

MR. BAJOREK: Okay, that's it, and also you're looking at a two-phase flow in a very transient situation, okay.

MEMBER MARCH-LEUBA: You are looking at a code that is expanding and maybe bowing, and it's something you have never done. Can you --

MR. BAJOREK: And more than one design.

MEMBER MARCH-LEUBA: And I'm with you. I'm a physicist. I'm a code guy. I would love to be working on this. I just don't think you can do it in 18 months. And if industry goes with this, you have to have feedback. I mean, you cannot take 18 months to do your computer metrics if you want to write an SER. So that's why I keep pushing the steady state codes, simplify volume assumptions, and on your slide 15, I would love to see PATHS, P-A-T-H-S, in there because that would be my main time for design. If in a steady state I cannot bound the results, return the design to sender because we cannot, honest, I don't think we can do it.

MR. KELLY: This is to show how by using the unstructured final and mesh you can actually model the geometry that we missed. On the left-hand side is HTR-10. If you look at the bottom of the conus, you see the perforated plate. That's the red region in the PRONGHORN model. Treat that as a porous body, the same way we're going to treat the porous medium for bent part.

Out in the reflector, these large holes are actually the upcomer. That's where the cold helium flows up through the reflector before it turns and goes down through the core. The small holes are the control rod drives. They're also cooled.

By using this unstructured mesh -- well, let me back up. The flow solution in those is going to be 1D, so it would be 1D flow channels coupled into the 3D conduction mesh, but by being able to model this, we don't have to make assumptions about, well, what's the thermal resistance from the edge of the core out to the edge of the vessel? We can actually do the conduction heat transfer because we can mesh it.

MEMBER REMPE: So since you brought up, I know from the last discussion we had on this topic, someone from DOE said, "Hey, we don't really -- we use it for designing the experiments, but for authorizing the safety case, they do not use it." So again, I'm not sure if that's a good reference for pedigree here. It's just a design tool.

MR. KELLY: But that's better than not used at all, which is the perception that these codes are brand-new and these are first use. So that one's actually been in use for a while and there are some capabilities in this one and the next slide that speak to that.

I don't have time to explain what advanced equivalence methods are, but it's a way of going through from a 3D Monte Carlo solution where you get not only the macro cross sections, which you would think about using a diffusion code, but you get a reference solution. You use that reference solution to get what are called SDH factors so that you can --

MEMBER CORRADINI: Is that a nice word for fudge factors?

MR. KELLY: In effect. If you will, it's a micro resolution of the cross sections.

MEMBER CORRADINI: I mean -- okay, fine.

MR. KELLY: And at the state points for which you calculate them, they give you a great solution as you'll see here. Between state points when you're interpolating, just as you do the cross sections, that's where we have to check the accuracy as we go through it.

So these are just two examples. The SERPENT number gives you the eigenvalue, and the plus or minus here is the statistical uncertainty in the eigenvalue. The yellow line is diffusion. The green is diffusion with SPH. SPH nails the eigenvalue at that particular state point as it should.

The last two columns are the RMS and maximum error. On this one, which is for the HTR-PM, which is a pebble bed, it's the neutron production rate that's being compared, and on the HTTR, which is a prismatic gap coolant test reactor, it's the top.

But the point of this is you can get transfer level accuracy for the computational cost of diffusion. That's what makes it amenable to us. If we had to do full transport on a core, it's not happening.

MEMBER CORRADINI: So --

MR. KELLY: And --

MEMBER CORRADINI: Go ahead.

MR. KELLY: The multi-scheme capability comes from tree, and what it allows you to do is run two different schemes at the same time. So for example, in this SFR, here is the control rod cluster in the middle.

You can treat the area just around that control rod cluster with full transport, and then all of the rest of the core with diffusion, diffusion with SPH, and that way you get the detail where you need it, which in the tree example, would be inside the test capsule, but you get to use diffusion for all the rest of the core which makes you able to put detail if you need it. You don't do it just for the fun of it, but if you need it, you have the capability there.

That's a calculation I did. SAM has the capability in it for a 3D flow model. SAM is really two codes in one. Its initial purpose was to be a modern equivalent to the systems analysis capability in the SASS array code. So it's a system level, component-based code.

MEMBER CORRADINI: This is a replacement for the old Brookhaven SSC code that coupled to SASS, wasn't it? I mean, that's what --

MR. KELLY: Yes.

MEMBER CORRADINI: -- the super system code was many --

MR. KELLY: Well, not super system. It's a replacement for the SASS model inside of the SASS array.

MEMBER CORRADINI: Oh, it's the system tool though and not the core model?

MR. KELLY: It's a system, but also the core, but the core only up to the point of damage.

MEMBER CORRADINI: Okay, all right.

MR. KELLY: Okay, it doesn't handle sodium boil, for example, single fast. So it's component based like normal or as fuel systems go. It's like you say a pipe goes from here to there, and it measures it up, and you link the pipes as so on, but it also has the capability to handle 3D unstructured meshes, and we use that in two different ways.

One is if you want to call it a CFD-light.

So it's like a 3D flow solution for a region that doesn't have a lot of structure where we can't use something like TRACE. So it has the 3D stress tensor, but the turbulence model in it's going to be algebraic, so it's much less -- reduced order is the way Argonne classifies it.

MEMBER CORRADINI: So it's just the diffusivity?

MR. KELLY: In effect, yeah.

MEMBER CORRADINI: Okay.

MR. KELLY: And so you might use this in our model where the core is a fast molten salt-fueled reactor, something like the Terrapower design rather than trying to do CFD for the core. That would, you know, not -- that's a level of detail we don't want to go to.

But it was initially put in for pools in pool-type SFRs where it affects the temperature that the intermediate and the decay heat removal heat exchanges see in the pool.

The other way is what I've already showed for PRONGHORN which is 3D solids with embedded 1D flow channels, and that would be something like a molten salt fueled thermal reactor, the design of Terrestrial.

And the way this all works together is the 1D components would be in one application, if you will. The 3D mesh would be in another, and they run concurrently using the MOOSE multi-app system.

In fact, you would typically, a lot of the different pieces of the system, you would have different apps for, and each one would be parallel in and of itself, and the way MOOSE is structured is to make it easy for all of these things to communicate and run simultaneously.

MEMBER CORRADINI: And then -- so that once again the transfer from this tool kit to the MELCORE tool kit would be based on some temperature or strain criterion where I'd expect fuel failures that this can't handle?

MR. KELLY: Exactly.

MEMBER CORRADINI: Okay.

MR. KELLY: And the last slide is a couple of the very simple validation problems. Remember, this is early days. So one is natural convection in a square cavity. The other is a lid-driven cavity flow. And these were compared to experimental results and it basically means you've got your sheer stress tense for volume.

CHAIR BLEY: Joe, I made a facetious

comment earlier and you gave a very thorough presentation, but I made the comment seriously. From your experience in getting into this, if somebody outside decides to use MOOSE in these coupled codes, is there any chance of them thinking it's too easy and not really doing the things you need to do to solve these problems correctly?

MR. KELLY: Well, first, I'm --

CHAIR BLEY: I mean, you'll be reviewing them here, but go ahead.

MR. KELLY: First, I'm very glad you brought this back up because what I forgot to say is that I have been very fortunate to have the help of staff members at Idaho National Lab to help me learn to use this code.

It wasn't like I picked up a code manual and got up in my loft and, you know, oh, yeah, this is easy. No, I have on, I want to say four occasions now, gone to the lab for a week and they set me up in an office. They give me a computer and I sit there, and there's a dedicated staff member or two.

So I will work through tutorial problems or a problem I bring with me, and when I get stuck, "Help," and they come and work through it with me, and that collaboration has been very good, and without that, I wouldn't have made much progress at all.

So while they are easier to learn than you might think, it's not easy. I mean, first off, we're talking about reactors, coupling reactor physics to thermal fluids, to structural mechanics. That's kind of a brave new world for all of us.

CHAIR BLEY: Thanks.

MEMBER MARCH-LEUBA: Yeah, and how do you validate the code? After you have a model for the plant, how do you know you've got it right? Go for it.

MR. BAJOREK: We don't yet. I mean, and we'll talk a little bit on our approach on how we get there, and that comes in some of the gaps. I mean, when we had our meeting, you know, that's what we thought we heard. You wanted to see what the gaps were, and a large part of those come through verification and validation.

In the interests of time, I'm going to skip through a couple of these things. With verification, hey, we have this idea. We have coupled codes. We need to have some additional cases to make sure, as one set of information goes from one code to the next, to the next, to the next, we conserve mass energy and momentum.

I'm sensitive of that because I was working on the coupling of COBRA and TRAC a number of years ago, and that's where we had one of our greatest difficulties.

But validation, some has been completed. A lot of it's ongoing and planned, okay, and what we're doing right now is we're identifying what that necessary V&V tasks are and we're putting together another report that's going to summarize all of this work that needs to be done. The approach --

MEMBER REMPE: I'm sorry to interrupt you, but didn't somewhere I read in your document that some of these codes also need manuals basically or user guides to --

MR. BAJOREK: Yeah, yeah, yeah, documentation is the one thing nobody likes to do.

But our approach for the validation is first to take what we've learned from the PIRTs, identify those gaps, okay, and review the existing information out there, the HTR-10, the MSRE, the EBR-II, and organize that data in order to get those validations started, and prioritize those tests based on the physics which are most difficult for us to do and the need by NRO to address some of these plants.

To get really all of them at the same

time, that is not going to happen, okay. We are going to need some prioritization to focus on this design or this type, and if we have that guidance, we'll be able to make the 18 months in doing all of this, but it's going to mean we're going to have to focus on one or the other.

That's also why we're developing what we call a reference plan. It looks like, smells like, should behave a lot like the plants as we think they're going to come, but we don't have all of the proprietary information.

It's just sort of something that looks like it, but this gives us a way of taking these coupled codes, taking them for a test drive, making sure that they work. If there's a problem, we'll get those fixed now before the application and saving us that time.

We'll also be able to exercise and then hopefully identify is there the type of margin that we think there's going to be there or are we starting to butt into some potential regulatory framework that we need to concern about, and focus on those things that drive those uncertainties. That will help save us time as that comes in.

As I mentioned before, we're working with the Department of Energy to split up the load. They'll be doing of the validation. We'll be doing some of that. We've started some of that ourself with the CEFR, for neutronics, the FFTF, okay, as a way of learning these codes and getting this thing going.

Validation status, it's, like I said, there are some that has been completed. It will likely need to be repeated at some point as we get to a frozen code for a certain design type, okay, but you get those input decks done and ready, and with ways of rerunning lots of decks. That's becoming a lot easier to do than it was a number of years ago.

The validation is partial. We realize that and we're going to be working on a plan to get that accomplished here in the near future. More importantly, there are some experimental gaps. We've identified things that we want to validate based on the data we know of and we can have access to.

In some cases, there are going to be problems, gas-cooled prismatic reactors. HTTR has been down since Fukushima. The OSU facility has been having problems with their heater rods. There's no information that's coming forth for a gas-cooled prismatic. SFRs and liquid metal reactors, there's a very limited amount of integral test data.

And if you go down to the bottom in the yellow, the idea is that no one is going to rebuild EBR-II. It's going to be a design that has potentially much more power or has other safety systems. It will certainly go to a pool type reactor rather than a loop type reactor.

So it remains on all of us, both industry and the NRC, to convince ourselves that the data from these existing tests scale and have the appropriate range of conditions for the new designs which are coming in.

Heat pipe reactors, there's not a lot of information on having some type of a monolith conduction test. There is some information out there. In MSRs, we rely on the MSRE, another loop type system, but there's nothing out there that will look like the pool type reactors that the applicants are talking to us about.

So this idea of experimental gaps goes twofold. There's validation that we still need to do with the existing data. We need to qualify that data, and there's going to likely be a need for additional experimental data for us to validate our codes and for industry to move forward on some of these designs.

We've already covered this one, so I'm

going to jump here to Dr. Bley's favorite slide here on the PCMM. We're trying to use that universally. We have it in volume one. We've got it in volume three. It's our way of tracking where these gaps are.

We've just started this. What you see there right now is just an initial shot at where we think this is, but as Kim pointed out, this is a work in progress, and as we move forward on some of these designs, we hope that our opinion and ratings of these various categories and elements improve over the next few years and we get to the point where both we and the user office are satisfied that these tools are giving the satisfactory accuracy for the safety case.

So to kind of close on this, our strong recommendation is that we go with the codes in this CRAB suite as a way of addressing the non-LWRs. There are gaps and there's a significant amount of work to be done.

I think that even though there's a lot of work to be done, we can make those review schedules. I think there is flexibility in what the tools can do and we've demonstrated to ourselves we can get up to speed in using these tools in a relatively short period of time.

CHAIR BLEY: A few quick things. One, is

there anybody here who takes credit for the acronym? And two --

MR. BAJOREK: DOE has this thing about naming after critters, and we can play that game too. The other thing would have been the Cranky --

CHAIR BLEY: It has two meanings.

MR. BAJOREK: CR could have been Cranky Regulator Analysis Bundle or continuing resolution.

CHAIR BLEY: The thing Mike asked you about before, envisioning your map of CRAB, somewhere along the line, you need to develop, maybe you've already done this, but something that would essentially be guidance for reviewers of which they ought to apply when, you know, and at least at the levels in this report, I don't think that's there yet. Have you played with that at all or you just know how you would do it and --

MR. BAJOREK: Well, I think that's called the Standard Review Plan, but unfortunately, that doesn't really exist for some of these design types.

MEMBER CORRADINI: But I think what he's asking is -- I think what Dennis is asking connects to what Jose is asking which is if you actually had to do this with a submittal, how are you going to go about doing it? How are you going to train the staff?

Forget about writing it down in a procedural fashion, but at least the steps that it will take that you need to go through so that you could redo what Joe has learned by going out to Idaho a week five times. You can't do that for all of the staff. Maybe you can, but I really do --

I really am concerned and I think Jose brought it up and Dennis is kind of coming back to it, some sort of organized process that gets the human involved because the engineer that has to deal with this is your critical link.

If it were me, I'd want to have it as simple as possible, as little data as possible, a few key parameters that I'm going to check so I know what I do in the world of LBEs that are design basis, and then I pass it onto beyond design basis folks with a different tool set so that it's very clear.

Otherwise, it could be, I don't want to use morass, but it could be complex, and I think that's what I sense Dennis and Jose were getting at.

MEMBER MARCH-LEUBA: I will not know until two months into the year.

MEMBER REMPE: Well, I think it would be hard to do that unless you have a design, so maybe your, whatever design things you're coming up with your standard plans would be a good place to try and do this and develop it. Because when I think about what you're suggesting, it sounds good, but jeepers --

MEMBER CORRADINI: Well, that is in my last attribute that I wrote down myself is phase development. You're going to have to make a decision of which one of these are you going to see first, take that as a pilot, and then decide the process with that pilot.

Otherwise, you have -- I count five -- you've got 10, but I will count sodium, a liquid metal, a gas, thermal, a solid fuel. There's too many.

MR. BAJOREK: I know.

MEMBER CORRADINI: It's just too much.

MR. BAJOREK: Yeah, the --

MEMBER CORRADINI: So you're going to have to have some sort of phase development that then tacks through how you arrange all of this in a process pamphlet with the analyst engineer, whether it be onsite or by contractor, to think it through.

MR. BAJOREK: Our initial step right now is with those reference plans. The ones that have filed this RIS basically, they look like they're leading the marathon at mile two.

CHAIR BLEY: So you're building reference plans for all of those.

MR. BAJOREK: Eventually we would like to do that, but for the ones that look like they may be leading the pack this way, we're developing reference models for at least three of them, and we hope to have those this year.

CHAIR BLEY: Okay.

MR. BAJOREK: And as soon as we have those --

CHAIR BLEY: That will help us understand where you're headed.

MR. BAJOREK: Then we start to exercise those, and then other ones will follow after that. And once we start to exercise those for the scenarios that have yet to be defined for us, we're going to be able to start sorting out, "Hey, is this a design basis activity where we need to assure there's margin for this certain type of event that goes on," or, "Sorry, guys, there's fission products going out the window and this is something that we have to treat as a severe accident." I think we're still a year away before we get to that point.

CHAIR BLEY: Thank you.

MEMBER REMPE: So for the three that

you're doing the reference plan, I assume they've also put in a Regulatory Engagement Plan or they just responded to the RIS?

MR. BAJOREK: I think it's both, but I'm not sure about the Regulatory Engagement Plan.

MEMBER REMPE: And then have any of them said they had tools for modeling and simulation or do they seem to be concrete, I guess, is what I'm trying to get? Have they thought about how they're going to do their evaluation?

MR. BAJOREK: Yeah, at least two of them have talked to us about some of the tools.

MEMBER REMPE: Are either of those going to use the DOE package?

MS. CUBBAGE: You would have to refer back to the presentations from --

(Simultaneous speaking.)

MS. CUBBAGE: Amy Cubbage, NRO staff. Refer back to the presentations from ACRS.

MEMBER REMPE: But I don't know which one are the three, I guess, so that's why I'm curious, and you probably can't say it here.

MS. CUBBAGE: Oh, I can tell you the ones that have submitted regulatory issue summaries. Those were the ones on Kim's slide.

MEMBER REMPE: Right, and he said there were three that have, are going to be starting with reference plans.

MS. CUBBAGE: Oh, so let's divorce those two concepts. The reference plans are generic, nonspecific, nonproprietary ideas of a design to build our capability. Separately, we have designers that are building specific designs.

MEMBER REMPE: Okay.

MS. CUBBAGE: Yeah.

MEMBER REMPE: Are they gas? Are they -- can you tell us what the reference plants will be? What kind of ones are they?

MS. CUBBAGE: Well, the reference plants that Steve was talking about are one of everything, a gas reactor, a micro reactor --

MR. BAJOREK: Yeah, we are developing one that looks like a heat pipe cooled micro reactor, okay. It's generic in its geometry and system --

MS. CUBBAGE: Informed by work done by DOE and NASA.

MR. BAJOREK: Yeah, there are other things out there that look a lot like what the designs are. We're using that information to put one together. We're putting together one for a pebble bed gas-cooled reactor.

I don't know what X-Energy has completely, but we know it's in the HTR-PM, okay, and we can take that and scale that up to get at power levels which are more consistent with where the applicants are going.

We are also in the process of looking at another one of those designs that involves a molten salt, but I don't want to mention which one that is because I don't know what's proprietary or nonproprietary, in order to get ready for another one of those designs that are out there in the blue.

MEMBER REMPE: And the molten salt has stationary or moving fuel?

MR. BAJOREK: It has fuel.

MEMBER REMPE: Okay, thank you.

MEMBER CORRADINI: But we had, in November, just to get to Amy's point, we had in November from the --

MS. CUBBAGE: You heard from X-Energy.

MEMBER CORRADINI: We heard from Framatome, Kairos, NuScale, Oklo, Westinghouse, X-Energy.

MS. CUBBAGE: Yes.

MEMBER CORRADINI: And they had specific

tools that they identified, or at least levels of tools they've identified that they would or would not choose to use as their design versus their transient analysis tools.

MS. CUBBAGE: Right, there's different levels of progression in where the potential applicants are in their code selection and their code development, and the details of what we've heard from them would be proprietary.

MEMBER CORRADINI: And then we also -- I forgot one. The technical working group for the liquid fuel world had their own presentation.

CHAIR BLEY: Are you finished?

MR. BAJOREK: I was going to ask you that.

CHAIR BLEY: I think we're finished. We're going to take a --

MEMBER MARCH-LEUBA: A quick 20 seconds.

CHAIR BLEY: 20 seconds, go ahead.

MEMBER MARCH-LEUBA: I wanted to reemphasize what they said before on the CRAB system. I see missing a steady state core simulator, so if we wanted to simulate our MELCOR, and we have PATHS.

MR. BAJOREK: Well, I'm not sure you'd want to use PATHS for all of these reactors.

MEMBER MARCH-LEUBA: I'm telling you

something like that like.

MR. BAJOREK: Okay, something like that, but as we do with TRACE, we set the model and we use that code to simulate the steady state.

MEMBER MARCH-LEUBA: And someone has done that?

MR. BAJOREK: I'm sorry?

MEMBER MARCH-LEUBA: The staff is not doing it for the reactor that we're reviewing now. They're not doing the steady state calculations very effectively. They have to rely on the vendor to do those calculations.

MR. BAJOREK: Oh.

MEMBER MARCH-LEUBA: It is clearly missing.

MR. BAJOREK: I think we would treat everything as an evaluation model where it's our package of codes that comes in and does the calculation.

MEMBER MARCH-LEUBA: It would simplify your evaluation.

MR. BAJOREK: Okay.

MR. BAJOREK: And I wanted to -- I do appreciate the questions and really I wanted to thank everyone for your attention.

MS. WEBBER: And I just want to echo that. I mean, it's a very critical review of our reports and we really do appreciate your insights and your comments. I think it will help us go forward.

CHAIR BLEY: We're going to recess until 20 till. We're going to start promptly at 20 till 4:00.

(Whereupon, the above-entitled matter went off the record at 3:20 p.m. and resumed at 3:37 p.m.)

CHAIR BLEY: We are back in session and I'll ask Hossein Esmaili to continue on.

MR. ESMAILI: All right, thank you very much for inviting us, and I'm going to go through our plan. Although I'm presenting this plan, I just want to emphasize the point that this is a team effort. We are taking advantage of our subject matter experts in my branch and also the scientists at Sandia and Oak Ridge National Laboratories.

When it comes to thermal chemistry and fission product, of course we are still relying heavily on the work that Dr. Dana Powers has done on how we can leverage some of those work for our non-LWR work.

I want to say that although we have some details in our plan, at the high level, I'm hoping to make the case here that our plan is the best, and using our available tools is the best approach, and from what I'm hearing before the break, I just wanted to make that point.

So I will say a few words about source term here. This is -- the importance of regulatory source term is well established. It winds its ways through a lot of our regulations for light water reactors. This is both safety and environmental reviews.

So this slide shows the development process for the design basis source term. This is an example of what was done for MOX and high burnup fuel. This is essentially the same process that was followed in NUREG-1465 and Reg Guide 1.183.

It was here adopted for high burnup and MOX fuel. It starts with identification of phenomena that's important, the experimental basis, and how these experimentals and phenomena are integrated into our code.

For this particular case when we are talking about MOX and high burnup fuel, of course we need data, and this data comes in terms of -- data comes from the facility of cesium for high burnup and MOX fuel rods.

The next step is identification of accident scenarios. So here I have the accident scenarios. These are informed by PRA as well done back in NUREG-1150. That was the input to 1465. And for defining a source term, then we synthesize these release fractions. So we care about the timing, release fraction, and how these source term are going to be used for licensing and for our license.

Okay, so now in this picture, I'm trying to put together what we know about light water reactor and how I'm going to go from light water reactor to these different technologies because I have one code. I'm not going to get into the code right now, but I just want to talk about phenomenology, release paths, et cetera.

So this is to identify similarities and differences between various technologies and how best we can approach this.

So at the very, very top level, you see there are phenomena that's happening in the containment. This is the evolution of condensation and deposition, resuspension, evaporation, some vapor and fission products, and aerosols in the containment, agglomeration and different deposition mechanisms, and how things get out of the containment and into the environment.

So at this level, once things get into the containment, we are basically, it's technology neutral. So once aerosols and vapors get into the containment, I can use the existing methodology that I have available.

There are going to be some differences here if I look at light water reactor versus high temperature gas reactors. The evolution of aerosols are going to be under steam conditions, for example, for light water reactors. It's going to be dry in HTGR, and so we need to take into account shake factors, et cetera, that's going to be different.

DR. PETTI: Just a question. The other thing you have to think about in physics models is the magnitude. The amount of fission products coming out of each of these systems is vastly different, so much so that physical models for condensation, deposition, and chem resorption, for instance, in a light water reactor will not be applicable in other systems because the concentrations are so low, it's a different physics regime.

MEMBER CORRADINI: Because you just literally, what would dominate in a light water reactor doesn't dominate in the other regimes?

DR. PETTI: Right, for instance, you can't form an aerosol in a gas reactor. In a light water reactor, you can form an aerosol because there is so much stuff coming off the core, it will form its own aerosol. You can have graphite dust as an aerosol and a deposition onto it, but the deposition models assume bulk phases, bulk properties.

The posture, the equilibrium posture pressure in a gas reactor for some of these species is 10 to the minus 18 atmospheres, for instance. That's a different physics regime than a light water reactor where we've got kilograms of material coming off the core.

MR. ESMAILI: And I think we had this discussion about 10 years ago.

DR. PETTI: Probably, yes.

MR. ESMAILI: And we showed you what happens not only in the containment, but also in the circuit.

If during steady state operation, if you don't have a lot of fission products coming in, and you were talking about circulating activity, the concentration of the fission products are going to be so low that we are going to reach a steady state.

So what's going to happen is that those

fission products are going to deposit on dust particles, et cetera. They're going to deposit and then they can get reborn and, you know.

So at some point, we still have to work with the aerosol particles because they're going to attach themselves to the dust particles, but I'm just drawing the similarities between different -- and as I said, yes, there are differences.

So if I go to the next levels here, you see the blue. This is what's happening in the primary system, so of course we are expecting differences here.

The way that fission products evolve and get released from fuel rods in a light water reactor is going to be different than TRISO fuel particles, but even here, the underlying physical physics are the same.

There are still driven by the diffusion process through these various layers, and so we need that to do our model. So whether we are using a coarser boot model, for example, or a simplified model for LWR versus I have to solve a diffusion equation for HTGR, the physics are the same.

The other -- I think we talked about some of the other mechanisms that can happen in the primary system in terms of aerosol formation, condensation, et cetera, that might be different in different technologies. If you go at the lower level, now here things are going to start to change.

In light water reactors, when things go ex-vessel, there are ex-vessel phenomena that are going to be completely different. For example, for a light water reactor, I have fission product released from the core-concrete interaction, okay. I don't have this in others.

There is some synergy here with SFRs because I could possibly have interaction of sodium with the concrete, but here we are treating the models differently, and we do need to have dedicated models for this phenomenon.

My talking points were very short. I was expecting to be interrupted, so I just get going, so maybe we can be done by 4:00.

MEMBER KIRCHNER: Can you go back one slide? Go back one slide now if you would, please.

MR. ESMAILI: Do you want me to go back? Okay.

MEMBER KIRCHNER: Okay, the slide that follows is your synthesized timing and release fraction, but explain in the second bullet down there under the imminent Dana Powers.

MR. ESMAILI: This one?

MEMBER KIRCHNER: No, the next one.

MR. ESMAILI: Differences not from change of fuel, but from code?

MEMBER KIRCHNER: Yeah.

MR. ESMAILI: From code advances. So in the old days for NUREG-1465, I'm going to get to that a little bit later because --

MEMBER KIRCHNER: Okay.

MR. ESMAILI: -- we had --

MEMBER KIRCHNER: You glossed over this pretty quickly, so, all right.

MR. ESMAILI: So I waiting for --

MEMBER KIRCHNER: Okay.

MR. ESMAILI: -- people to interrupt.

MEMBER KIRCHNER: All right, yeah, it's not going to be that easy.

MR. ESMAILI: Okay, so in the old days, we had a source and code package. Most of the stuff that was done early on was done with dose models. I'm going to get to that later.

For this particular application, we have MELCOR, so the models became more and more mechanistic, more and more sophisticated through layers. But still then we are looking at the fractional releases, we are seeing that it doesn't change.

So, a good thing that you mentioned this because if you look at that cesium diffusivity, as the temperature goes higher as I'm going to the left of the figure, you know, whether it's a low burnup or a high burnup, they come together.

So in other words, as the temperature goes up and I'm releasing more and more of some of these volatiles, I'm releasing more of the same amount.

MEMBER KIRCHNER: When things go really bad, it comes all together.

MR. ESMAILI: That's right, for the volatiles, for the volatiles.

MEMBER KIRCHNER: Well, for the volatiles, yeah.

MR. ESMAILI: So the other thing is I want to show you this scatter in the data. This is on a log scale. I'm going to get to that a little bit later, but you see the type of data that we are getting when it comes to severe accidents.

MEMBER KIRCHNER: One thing, Hossein, that I know this is almost heresy. But we've learned a lot from LWRs. And when we make these comparisons, it seems to me MOX fuel is like LWR fuel. Like your next chart, you do have it. You have the LWR there. It's good just to -- as a reference point and the gap analysis, to just point out where the unknowns really are, the new -- where we're going to need data or whatever. Maybe you were going to come to that.

MR. ESMAILI: I'm going to -- I have this slide where the data needs are.

MEMBER KIRCHNER: Okay. Well, then --

(Simultaneous speaking.)

MR. ESMAILI: Right.

MEMBER KIRCHNER: Okay, good.

MR. ESMAILI: And I'm going to get into that a little bit later in terms of basically what we have in MELCOR is that we have a framework for a lot of this physical model. What's going to change some of the data that's going to come from the experiments, et cetera. Okay. So --

MEMBER KIRCHNER: But to Dave's point, how generically applicable are the models when some of the -- are you looking also when the chemistry and the phenomena that are actually driving the accident scenarios change? Do you expect MELCOR will need a change in models?

MR. ESMAILI: I -- well, I'm going to make a case later on because we have looked at sodium fire experiments. So I'm actually showing --

MEMBER KIRCHNER: That was one that was in my mind, yeah.

MR. ESMAILI: So this is under sodium fire. This is a sodium fire that's occurring, right?

MEMBER KIRCHNER: Right.

MR. ESMAILI: We do have some dedicated models to model sodium fire. But when it comes to the aerosols, I'm not changing anything. The math equations and the aerosol dynamic is part of the code I have not changed. And I'm getting reasonable results in terms of what I'm predicting in terms of airborne.

MEMBER KIRCHNER: Okay. All right. Okay.

MR. ESMAILI: So this is what I can say right now. So Slide 4 is that when it comes to the knowledge gaps and I've identified some of the reports, some of the PIRTs that has been done in the past for HTGR. These are the PIRT that was done as part of the NGNP. This is by no means a very comprehensive of set of references. Some of them are more formal PIRTs than others. But there is information available to guide us on what we need to do in articles. That's all I want to say here.

When it comes to -- so in terms of HTGR and SFR, we are further along in terms of PIRT and understanding the phenomenology, et cetera. For MSR, when it comes to fission product release and evolution from a molten salt, there are two PIRT meetings that are scheduled for this summer. So we're going to be participating in those PIRTs. And this would form the basis of what our modeling should be.

In the report, we are -- again, we are explaining that evolution of fission products from a molten salt is driven by vapors. This is actually by Dana, so we reached out to Dana. Vaporization from the molten salt and bubble bursts at the surface. This is the same type of phenomena that we expect in a corium, right?

And so our thinking is that let's go back and look at the VANESA model that we have, look at the fission products, how they get partitioned into the -- and how we can evolve and use that type of information so that the data has to be completely different. The type of fission products we have in a molten salt is going to be different than in a corium.

DR. PETTI: So just a question on VANESA and maybe something that Dana has to answer. How sophisticated are chemistry models to account for potential non-ideality? I mean, degradation is easy once I know the partial pressure. But knowing the partial pressure is where the chemistry gets interesting. Can you account for non-ideality? Can you account for impurities in the salt? I mean, I think the structure is right, but it may require a little bit more.

MR. ESMAILI: Absolutely. That's what I think we are working right now. As a matter of fact, one of our input is that these are the things that we are thinking about going into the PIRT process, and we're just going to see what we need to do. If we need to change a little bit here and there, we are going to do that. We do have -- okay, I'm going to get to that a little later.

All right. So Slide 5. So when it comes to HTGR development, these are the tables. I'm not going to go over this. I just want to make sure that these are based on what we reviewed as part of this PIRT, we came up with some key phenomena that we think is important that should be in the code.

These are, like, how do you model TRISO fuels? What is the heat transfer, graphite oxidation, modeling of the dust transport, et cetera, air-moisture ingress. Why they're important, of course, why they're important is that they came from the PIRTs. Do we have existing capabilities in current modeling and what are the modeling gaps?

It turns out that we have done a lot of work as part of this NGNP back ten years ago. So we could not identify a lot of modeling gaps. I just put in that their currently modeling uses UO2. If I just left that column completely white, you would say, are you serious?

So we have some modeling gaps and we're going to go back and look at some of this stuff. But the code is essentially ready to be exercised for HTGR application. And we find that more and more international users are using this for their applications. They can pick up the code. They can pick up the documentation and run the code for their particular applications. You want to ask?

MEMBER CORRADINI: I was going to ask. So if it's ready for usage, are you using the same generic pilot plan design that the DBA folks are using?

MR. ESMAILI: We developed an input model. This is the PBR 400 that they have. They have a prismatic input model and we have pebble bed input model. This was the work done as part of the NGNP.

We added some more balance of the plant into the mix. It's not a particular design. We have -- yeah?

MEMBER CORRADINI: Well, my question is, to put it bluntly, is your evaluation pilot the same as the evaluation pilots being used by the previous?

MR. ESMAILI: I don't know. It's probably not. Maybe -- I explained that this is a PBR 400, the one that we used.

MEMBER CORRADINI: Yeah, yeah. Sit at the chair's too.

MR. BAJOREK: Okay.

MEMBER CORRADINI: I'm just looking for consistency.

MR. BAJOREK: No, we don't have any consistency, not at this point. You're looking at a prismatic. We're looking at --

MR. ESMAILI: We're at the pebble bed.

MR. BAJOREK: We're looking at a pebble bed, just not the same one.

MR. ESMAILI: But if you have the design information, this is a good way of doing the same thing to just say that, can we get up to the time where we can potentially have fission product release? Can we have some --

MEMBER CORRADINI: But from the --

MR. ESMAILI: -- consistency --

MEMBER CORRADINI: -- standpoint of consistency, I would invite you all to talk to each other.

MR. ESMAILI: Yes.

MEMBER CORRADINI: Since you are, in theory, on the same tenth floor. Okay, good.

MR. ESMAILI: That's an example. This code to code comparison is benchmarking against some of the BDBA codes.

MEMBER CORRADINI: It would be fascinating to see -- I mean, well, if I'm into an analyst mode, it would be fascinating to see the same model plan with a detailed calculation -- I'll use the term from MELCOR, more simplified or a finite volume calculation and see the tendencies in terms of some exercise and some transients in where you go with it.

MR. ESMAILI: So I have in here -- I have in the appendix to our report, I just put in a presentation by one of our international users. This is the ALLEGRO model. So what they did is that they used MELCOR for they want to get fission product release, et cetera. But they also benchmark it against the DBA -- against the French code QATAR for DBA analysis.

They find out that they're satisfied with up to the point of code damage or fission product release. They were getting similar results. And this is in the presentation of the report.

CHAIR BLEY: Just -- you wanted to get in?

MR. CASE: Yeah, just a quick story. When we started to write development plans, I tried to force them to address the 13 generic things that was developed in the DBA area. As it turned out, there were a little more generic. So they stepped it back a little bit. So it's a good idea. We'll probably get there. We couldn't get there now. And so it's a fine idea, and we'll get there when more of the details come out.

MEMBER REMPE: Before you go to doing these comparisons, I liked your table in the report where you actually identified a lot of test experimental data like the Comity (phonetic) test, the AVR depressurization tests. And that's going to be done first before you go and do these other things in your plan. I didn't really see timing.

MR. ESMAILI: So I'm going to get to that a little bit later. Based on the resources and in the funding that we have available, what we decided is that for the next few years, we are going to build an infrastructure in the code which means that I'm going to be putting the models that are needed. We saw that for the sodium reactors because I need more modeling in the sodium reactors. Then I'm going to do the basic modeling for the MSR.

So I'm going to be doing some of this code infrastructure so that the basic physics, I'm going to put it so I can exercise the code. Once things become a little bit more clearer, past FY20, et cetera, we're going to be doing more assessment if it becomes available and testing of the code.

MEMBER REMPE: And then this plant and code comparison stuff would be even beyond the experimental test comparisons?

MR. ESMAILI: To tell you honestly, as I said, we do have international partners that are already running the code. We have people in Switzerland that are comparing our code results with HTR 10. So we can leverage some of that information, right, in terms of how we are going to do it.

So it's not going to be a serial thing. It's going to be we're going to do a lot of things in parallel. For Sandia to put a lot of models in the code, they have to make actual models because they have to test it. And sometimes when they make these models, it's possible to go and simplify just a little bit and see if they're experimental, if they're available, and do this.

MEMBER REMPE: I think that would be really good.

MR. ESMAILI: So these are -- yeah, yeah, these will be done in parallel.

MEMBER REMPE: Okay.

MR. ESMAILI: We have done it for the sodium. I'm going to go back and talk about sodium. We have done sodium fire and sodium pool fire and sodium spray fire experiments. So we're in our Volume 3 which is the code assessment report. So we have done that part of it as we were developing the model because we wanted to see the models that we imported from CONTAIN to MELCOR. Do they do what they do? So a lot of these things are done in parallel.

So this is the -- so here I don't want to say anything other than to say that we have reviewed the PIRT again. These are the key phenomena that we think it should be in the code for us to do a meaningful analysis.

So in the past, we only focused on sodium fire, sodium concrete interaction model. This was easy because the models have already been developed.

It was CONTAIN. All we had to do is support it into the MELCOR model.

We are doing some work on fission product and fuel degradation modeling. This is the work that we are doing this right now as we are getting funding. For vaporization, bubble transport, et cetera, these are all ongoing. So this FY19, they're really focusing on all the modeling needs for a sodium reactor.

Heat pipe thermal hydraulic and failure model, this, it says here, does not currently have a heat pipe model. But in reality, this was written some time ago. So we are at the point that we have exercised some of these development items. So we do -- we can write our model. So we are in the testing phase. So we have done the modeling of the heat pipe and we are just testing them right now to make sure everything works, et cetera.

All right. So for molten salt reactor, so this is the part that in terms of physical modeling, the equation of state for FLiBe and even other salt types, what we have done for sodium, important equation of state for sodium, is just basically the same. So it was generic enough so now we can model sodium -- sorry, molten salt with the same methodology as we have done for sodium.

So that is already in the code. This is already the existing capability. Heat transfer coefficients, there are heat transfer coefficients available. And as you know, we have lots of sensitivity parameters. So if need be, we can go back and change the heat transfer coefficients, et cetera.

For the track of flow of gases through the molten salt, we have a SPARC model. This is what we have in terms of bubbling of the gases through a water pool. And so we're going to leverage to see how we can do that for the molten salt.

These last two items is what I described before about VANESA model and how we are going to do it. And we are going to know these things more in the future as we interact with this as part of the PIRT process.

All right. So in terms of data needs, so I pulled together all the input and data requirements. Of course we are developing models. These are physical models, but these require data. And required data, so some of them come from experiments. Some of them come from the other codes.

Like, for example, for fission products, for fission product inventories in terms of the mass and activity, et cetera, it has to come from SCALE. And Don is going to talk a little bit about that later one.

But what I'm going to show at the bottom here, I just give an example of how it's done in the code. So at the very bottom, you see that I'm showing a picture of a TRISO fuel particle. So that equation, that diffusion equation describes the diffusion of cesium to these different layers. Okay?

So what is it that I don't know about this is the diffusions, that "D" in that term, that this comes from experiments-analysis. Okay? So at the time that we were doing NGNP, there were -- I don't know what the status of the AGR program is in terms of that. But I think at that time, they were trying to run the experiments with design to failure, TRISO particles, and use the existing PARFUME model. This is a 1D code INF to try to obtain the "D".

For us, it's an input. If I don't know this, I still can run it. I will just use the German data for our amounts. How these fission products build up in the kernel and the --

MEMBER CORRADINI: But wait, just slowing down for a minute. But to the extent that the experiments in AGR, the AGR experiments provide you with a back analysis of that, that that is then the experimental -- that's what you need for this.

MR. ESMAILI: That's right.

MEMBER CORRADINI: Am I understanding this correctly?

MR. ESMAILI: That's correct. So that "D" is an equation --- just like a model that is easy or exponential. So I just need two --

MEMBER CORRADINI: Coefficients.

MR. ESMAILI: -- coefficients to describe that. And that, we are expecting that to come from the AGR program. I'm just saying I don't need to make code changes. This is just done. It's really sensitive.

And so SCALE is going to provide us how these fission products build up. And this one shows up. What I show on the picture shows up. This is how the fission product -- this is Cesium-137 -- how the fission products build up in the kernel, how it diffuses to the buffer layer.

Because I have a silicon carbide layer, nothing gets out. So I know the information at the time if I want to start my transient. So if I have -- during the course of a transient, if I have a silicon carbide cracking or failure, et cetera, then I'm going to release whatever I have in that buffer layer and whatever comes off from the kernel becomes part of it. So that's why we do this pre-processing.

MEMBER REMPE: So somewhere in the report that you generated, I thought I saw a comment about on some of the MSR designs since they're going to use the pebble beds that you'd have data for doing that. And I was wondering, to me, when I think about graphite and molten salt and potential interactions, to me, I think you might not be able to just directly use that data. You may need additional data to think about any sort of interactions between these materials. But maybe someone who knows more about the reactor besides me should comment on that. But it came to mind.

MR. ESMAILI: So I'm thinking as long as there's TRISO particles in the pebbles, it's still driven by this equation.

MEMBER REMPE: The equation may be fine, but I think you need data. And what was concerning me was the report -- my quick reading of the report inferred that you thought you could just directly apply that data. And I'm not sure that's true. Plus --

MR. ESMAILI: Yeah.

MEMBER REMPE: -- the environment is

different --

MR. ESMAILI: That's right.

MEMBER REMPE: -- is what I'm trying to say.

MR. ESMAILI: So in HTGR, the fission products in case of failure of the silicon carbide layer, they still have to go through this matrix, right, of the pebble. And they have to do this fuel part, and then they get to the helium. And as we discussed, if it's very, very low, my circulating activity would be of the order of ten to -- I guess we showed that last time, ten to the -15. It's not even enough for me to form an aerosol particle.

In the case of the FHR to the molten salt, I would follow the same path except that the fission products get released from the pebble. They're going to make directly into the molten salt. That is where we are going to go and see what happens next.

DR. PETTI: So with that equation, you need more experiments.

MR. ESMAILI: So this is part of our -- that's right.

MEMBER REMPE: I think you need more experiments --

MR. ESMAILI: Yes, yes, yes.

MEMBER REMPE: -- because I think you might see earlier degradation is what I'm trying to say.

DR. PETTI: Because at least in the pebble bed, a salt will intrude into the pebble.

MEMBER REMPE: That's what I'm thinking.

DR. PETTI: Not in the prismatic because of the high density graphite probably won't, none of them using a prismatic salt. But the salt, the Chinese have done a lot of work on this. It will intrude into the pebble. And so my guess is you'll need some sort of a partition coefficient, an isotherm if you will --

MEMBER REMPE: That's what I think.

DR. PETTI: -- to describe what sits there -- what the bulk flow sees. And there's not been anything done on that.

MEMBER CORRADINI: So you would have to literally do an experiment on those particular conditions.

MR. ESMAILI: I think we have done some experiments. I don't want to talk about this, but this is at very early stages. The Canadian Nuclear Laboratories, they are actually taking salts. They're trying to find out how much of the fission products get out. But when you brought up isotherms, this is actually part of a prismatic. So when we go to the prismatic because you have this -- because of the way the geometry works, we have to worry about that, the discontinuity and how things get from the metrics to the graphite block. So we do model those isotherms for prismatic.

DR. PETTI: You may have to do something similar is what I'm saying.

MR. ESMAILI: That's what I'm saying. So we have that model and capability.

DR. PETTI: Just getting the data.

MR. ESMAILI: That's right. Getting the data, yes.

DR. PETTI: It's all about the data.

MR. ESMAILI: It's all about the data. It's just the physical model, I think we have most of it in the code.

MEMBER REMPE: And it's so much easier if you don't have the data. Things can be safer a lot of times.

MR. ESMAILI: Yes. Okay. So I hope we made the point that why. So I think the members were asking this question of what level of detail do you want. And so we thought that for the type of analysis that we need to do, we are going to rely on MELCOR. This is the code that was developed at Sandia National Lab for the past 30, 40 years.

It's the state-of-the-art tool, I guess, state-of-the-practice tool when it comes to non-LWR until we find out getting more data, et cetera. But the point I want to make is that MELCOR is already a code. Before MELCOR was designed, we had Source Term Code Package. This was a collection of codes that were run independently. The data was transferred between them and coming up with the source term.

So what MELCOR tried to do was put all of these separate codes into one code. So, like, for example, the SPARC model -- I talk about the core-concrete interaction, the VANESA model. These were all separate codes. So MELCOR made sure that all of them have the same database, et cetera.

So we were asked about what do other codes have. We looked at some of these codes that used. MAAP is used the industry. ASTEC, this is a French code. This is a severe accident code. They do not have all the non-LWR capabilities of MELCOR. So MELCOR was the right choice for us.

When it comes to ASTEC, they do have a version of ASTEC. It's called ASTEC-Na, so it's a sodium ASTEC. I don't know where the development on that one is. But once we develop our models, we plan to do some code-to-code comparisons just to -- you know.

So at the bottom, you see the timeline of what we have been doing. So if the process start in 2007, HTGR model, we were introducing this into the code as part of the NGNP program. Then NGNP stop in 2011-2012 time frame. So we didn't do any more activities on that. And now we are picking it up, and just took us some time to learn what's in the code and what we have to do.

Then you see the green line here. It says, sodium fire models, that went from 2013 to 2018. This is actually the sodium fire model. This is something that NRC, we didn't pay for it. This was done on the DOE side because these models as I told you were already in CONTAIN. For knowledge management, they just wanted to make sure that they wouldn't report it into the MELCOR because we are not doing any more development of CONTAIN.

And SFR models, MSR models, as I said, during the next two years, we are going to build this infrastructure of these models that are needed. And beyond FY20, this is a living document. So we're going to update what we need to do in the years beyond. We think it's going to be more on specific design and assessments, et cetera, as details and funding become available.

Okay. So I just want a review of some of these. I don't want to forget. So this, I just want to point out a few things from this slide. First of all, they have a large database. So at the bottom, you see there are a lot of countries that are participating in our program.

We have about 1,000 users. We have nearly 30 countries that are participating in our severe accident programs. And some of these users, as I show, there are European programs that are participating. There are people who are doing MSR. There are people who are doing HTGR. There are people who are doing sodium work. And as a matter of fact, someone asked us, when is the code ready because we want to do some of these analyses.

Some of them have already done the HTGR. Examples are people in Hungary and PSI in Switzerland that they're already using it. And more and more people, I think, are going to be using it for MSR as we build the models.

So on the right-hand side at the top, this is code development and regulatory application. We rely on the experiments. We do experiments to understand the phenomena and validate our model. So whatever you see in the blue is the phenomenology and how we use that. This goes into the code. And on the right-hand side, you see a large list of regulatory applications. So this is the process that we follow in terms of what we need.

So if you're asking, why are you using MELCOR, then we say, okay, we have used MELCOR for past regulatory applications. Staff and other code users are already familiar with the code. So extension to non-LWR is required minimal training. In other words, I can use the code for HTGR but I only need to just change the volume. I already understand my input requirements, for example, for aerosol dynamics. I may change them, but I understand what I need to do there.

And the interface between MELCOR and MACCS, we have done this over the past 30, 40 years. And when it comes to the PRA for non-LWR, the consequence analysis becomes an integral part of this analysis. So we've already established that.

And the code also provides us great flexibility for uncertainty analysis. So this is becoming more and more of a standard practice. We've seen SOARCA analysis, but we really need to do lots of calculations to see where we are.

Before I move on, I just want to make a very important point. And I alluded to that in my slide on cesium diffusivity that came from the experiments and the scattering the data. So I want to make the point. So when it comes to severe accidents, the uncertainties in accident progression -- this is uncertainties in accident progression. Uncertainties in boundary conditions, how things work. How does the valve work, et cetera.

And also available experimental data, and I'm referring to the scatter for model validation does not support higher-fidelity modeling approach. So I am still limited by what my experiment is telling me, what the rest of the system is telling me, how does this valve, how does it get stuck. This is what we have seen in LWR, et cetera. So those may be more important than anything that I put into the code in terms of uncertainty. So I just wanted to make that one clear. Okay. So is this a picture that I stole from NGNP.

MEMBER CORRADINI: I knew you stole it from somewhere.

MR. ESMAILI: It would not be stealing if you help develop it.

(Laughter.)

MEMBER CORRADINI: They had a reference.

MR. ESMAILI: That's right. So I'm going to go -- I'm going to walk through this. So when it comes to -- I put in on purpose live water reactors because this is what we do and this is what we have been doing for reactors, spent fuel pool for the past 30 years. I need to get these parameters, kinetic parameters, power distribution, decay, isotopic fission product inventories from the reactor. We're already there. We have done this work.

Then those are the input that goes into MELCOR. And what comes out of the MELCOR accident progression and source term, that becomes the input into MACCS which Jon is going to want to talk to. So that part of it is for LWR. We have already established, so that connection is there. When it comes to SFR, we are following the same path because these are stationary fluid rods and so we can get -- we don't have to deviate from this approach.

So when it becomes for HTGRs and FHRs, when we have TRISO fuel particles, we have this additional step. And this is what we also showed back in 2011 is that I need to do some fission product pre-processing. This is because I'm going to have some circulation activity that's going on. I'm going to have fission dust generation. I'm going to have some of these fission products attaching to the dust and depositing. But this would be my initial conditions for an accident. So I have to do --

MEMBER CORRADINI: How well do I know these initial conditions?

MR. ESMAILI: We're running the code. So in other words --

MEMBER CORRADINI: No, no, no, no, no, no. God bless the code.

(Laughter.)

MEMBER CORRADINI: It isn't what I'm after. I'm after it's a fabrication uncertainty, right? It's essentially manufacturing specs that I have to have some to start with.

MR. ESMAILI: So I would start off -- during the one operation, I would say that what is the failure of the fission of the silicon carbide? Let's put it this way. There's a very, very small percentage during normal operation that actually have failed, right? But I'm still going through the same process of how I'm passing fission products into the primary system. When it comes to dust, I have to rely on available data. I don't have any mechanistic model for this.

MEMBER CORRADINI: But what I should be asking -- maybe I didn't ask my question right. I'm understanding the pre-processing need. What I'm trying to understand is, is there any database in terms of what the range of manufacturing specs you demand -- don't look at him -- and the dust? The specs I'm wanting. But to me, the dust strikes me as an unknown quantity.

So does that mean there's something that's going to be required of the applicant to have some sort of measurements of essentially pre-operational testing or measurement of sampling? How do I know what that is so I know what is reasonable and unreasonable to assume in a calculation?

MR. ESMAILI: I wouldn't -- I can --

DR. PETTI: I can tell you. There was a dust curve done, okay, specifically on dust. In the pebble bed, there was actually measurements conducted in the German reactors, size and distribution, really good stuff, perfect for input. You can use that. For prismatic --

MEMBER CORRADINI: This is on the German

--

DR. PETTI: Yes, the German ABI. In prismatic, they looked hard and couldn't find dust which what you might expect. So that's that. In the fabrication space, there are specifications on every defect level at 95 percent confidence which is what I think you would use in a safety assessment. But there's also actual values batched back that you could use as a best estimate.

MEMBER REMPE: What about property? It was more the prismatic?

DR. PETTI: Yes.

MEMBER REMPE: And you can test to look at depressurization and dust --

(Simultaneous speaking.)

DR. PETTI: Yes.

MEMBER REMPE: What do you they think they have? Is that not something --

DR. PETTI: Oh, yes, yes, yeah, yeah.

MEMBER REMPE: You identified Comity can be used in your --

DR. PETTI: So Comity did the dry testing and we did the wet test. And the wet test was --

MEMBER REMPE: But it's something that they could do --

(Simultaneous speaking.)

DR. PETTI: Yes, there's data there, yeah.

MEMBER CORRADINI: So there is data -- so what I was trying to get at, there is a data respect basis on which one can know what is reasonable and unreasonable to doing the process.

DR. PETTI: So for instance, on the testing, the AGI program has shown what the designer assumes is going to be the failure rate. You can use that in your safety analysis but what's actual practice in this huge margin. And it's still a question on how the designer will decide to use that margin. Will they change their specification and take some of that margin back? Because this is all about balancing all these different margins --

(Simultaneous speaking.)

DR. PETTI: Right. I mean, also it's an optimization.

MR. ESMAILI: So this has to come from the vendors, right? They have to tell us. As far as I'm concerned, I know ADR was, like, two kilograms per second of dust generation. Or I don't --

DR. PETTI: I just know -- all I know is how many of them, how many kilograms there were and what size it should be. But actual rate, it's just going to --

MEMBER CORRADINI: It's a post-test integral measurement.

DR. PETTI: Right. But you could assume it's a variable that equals --

MR. ESMAILI: So I have to translate it into a rate and introduce that dust into the system during this process. I don't let that dust -- whatever that site particle is, let it deposit on the piping and everything. So that when it comes to do the analysis, these things can lift off and go up. But this is the part that we're doing here.

MEMBER CORRADINI: So in the FHR, it would be different because of the chemistry, yes? I mean, I understand the manufacturing defect fracture might be the same because of -- but now I've got pebbles inside of a molten salt, so --

MR. ESMAILI: You are thinking of those things, yes.

MEMBER CORRADINI: Because I don't have now a deep pressurization accident and I don't have anything. I just basically got whatever is out there is dissolved within the salt.

MR. ESMAILI: And the chemistry is also very important. So the way some of the noble metals behave in a molten salt, they play it out. So they have salt in fission products. So it's going to be different. So those are the kind of details that we have to worry about. But we do have a path forward. We just make the data on how to best characterize those.

MEMBER CORRADINI: Thank you.

MR. ESMAILI: All right. So MSRs, these are liquid fuel molten salt. So what I'm saying they do another step. So you go to a steady-state initialization. So all this requires is that I have certain amount of fission product. So all I want is to run the code, run the whole plant, and start mixing until I come to a steady state.

So now I have the distribution of fission products. You already have models to transfer fission products just like we are doing in water. All this does is just distribute it throughout the whole system. And I think Steve showed you a picture of the other control volumes. Like, what do you call it? The filter, et cetera, this could be additional control volumes that we can model and we can see where the fission products go. And at the start of the accident, so we have an understanding of the fission products in the salt in the covered gas.

So in terms of validation, this is an important part of the quality assurance. It's Volume 3. So what we see on the left-hand side is Volume 3. This is our assessment documentation. This has lots of information on light water reactor applications. Here I just circled two things that we think are probably applicable. These are the analysis that we have done for light water reactors, not all of them but some of them in terms of containment.

So if you remember I showed you those initial -- I said what the type of phenomena that occurs in the containment may be even applicable to none of them once the fission products get into that. So we are leveraging those. And how do we know they are good? Because we have done of the sodium fire experiments that shows that the models that we have put into the code, even in the sodium environment, they do what's expected. We've got minimal changes into the aerosol physics. So as we go on, we are going to do additional assessments of the code. And again, right now our main focus is on code infrastructure.

So concluding remarks. So we are trying to leverage decades of model development and validation can be extended to non-LWRs. So we're trying to keep it simple, whatever we can do. What's the shortest distance between point A and point B. This, we think, is the most efficient approach where accident progression and source term analysis. For some technologies, I talked about HTGRs, the models are ready to be tested and people are actually testing these models.

The plan, this is still a living document. The plan will be updated as more experience is gained and as new information becomes available. So I would know a little bit more about MSRs maybe a year from now after we go through this PIRT and we would add these additional requirements into our plan. That's all I have to say. Thank you.

MR. ALGAMA: Good morning. I'm sorry. Good afternoon. The stress is getting to me. Good afternoon. Good afternoon. I work with Hossein in Dr. Richard Lee's branch, and today I'm wearing my SCALE hat.

In this presentation, I'm going to provide an overview of the SCALE code and where it is right now, what are the needs for the MELCOR, MACCS codes, our analysis approach, and our current understanding of daily gaps to both understand the phenomenology systems and for the code validation sensitivity in the analysis. This has been a collaborative effort between a number of teams, between the NRC, Oak Ridge, and Sandia. I'm just the point guy today.

So this slide, what is our general approach for these reactors is threefold. Overview of the phenomena important to licensing for accident progression. What is the validation basis for these phenomena, and what are the gaps and bounding analysis acceptable. Three, build upon existing tools, nuclear data and validation that currently exists.

This is an example of how SCALE fits within our regulatory infrastructure right now. SCALE is a modern multidisciplinary tool developed for the NRC and regularly reviewed since the 1980s -- sorry, 1970s and used to support MELCOR since the 1990s. SCALE covers many regulatory areas of the nuclear fuel cycle and is currently used throughout the agency, NRR, NRO, NMSS.

On the left-hand side, we have examples of validation data. In the center, the capabilities of SCALE that has been driven by the validation data. And the right-hand side are some examples of how SCALE has been used in related applications in the past. On the bottom of this slide, it shows how SCALE is used by the NRC tool such TRACE, PARCS, FAST, and MELCOR

MAX, how we initialize them.

SCALE also benefits from a large user database which includes foreign regulators. The takeaway is that the code has many opportunities to be exercised and pushed to the limits. So that's what we're looking for, for an LWR, is pushing the current infrastructure to the limit and see where we need to improve.

Focusing on this approach how SCALE will go to MELCOR, based on our work in the past, in this presentation, I'm going to give you an overview -- I'm sorry. How we use MELCOR/MACCS, let's start with the current analysis approach for LWRs. There are six general steps that we use. Three of them are assessments and three of them are actually used to generate the actual data used for MELCOR.

First step is accumulating and curating nuclear data including covariance data which is appropriate for that application. Based on experience from LWR work, there are long lead times. You take years of specialized experience to get the right data set and curated for use in applications. However, with SCALE, we've already done this and we have a nuclear data team that runs the AMPX code. So we are ready to go, and we have their library sets to work off.

Then we develop the appropriate methods. Then we validate those methods using decay heat and isotopics data for depletion and criticality benchmarks. Then we apply the validated codes to understand the impact of operating history such as power levels, control rod effects, et cetera.

Apply the validated codes, analyze the impact of specific accident scenarios to modeling details, and then use the validated code to develop the required data for MELCOR and MACCS. Upon preliminary review, these exact steps will be modified and used as a basis for LWRs.

MEMBER CORRADINI: So nothing substantial would be changed except for the data sources?

MR. ALGAMA: The data sources and then we would have to use other capabilities that have just come online for MSRs, for example.

MEMBER MARCH-LEUBA: I was going to ask about that and --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: -- TRISO fuel.

MR. ALGAMA: No, so we'd have to use TRITON MSR and then combine that with Thermochimica. It's on my slide to discuss. There's going to be a boundary between where SCALE ends to initialize MELCOR and where that goes. And that's a discussion that only just started. So we've just -- this is a plan and we're about to start the actual work.

MEMBER MARCH-LEUBA: Does TRITON MSR exist?

MR. ALGAMA: Yes, it'll be in SAIL 6.3. And Thermochimica exists too.

So this is a slide to tell you the diversity in the proposed reactor designs. In fact, I need two slides, one for here and the next slide. The data included here is not final and shouldn't be used as a reference. But just to show you just the spread of the reactor designs.

We can see that they are up to 20 percent enriched fuel, operate different neutron spectra, different types of coolants moving fuels through a different fuel form from metallic rods to TRISO to liquid fuels. Our approach then must be flexible and offer appropriate capability to capture these variations to meet the needs of accident progression. So this is another example of MSRs and the types of variations there are.

So now that we understand hopefully how we do things for LWR space, how we're going to move that to non-LWRs. In our SCALE report, we discuss all the different steps that will go to adapting our existing methodology.

To support accident progression, it is expected that many assessment cases will have to be performed to explore the parameter space to suit the needs of each reactor class based on that reactor class considerations. Some of those considerations are included here. They're highlighted. To support accident progression, it's been in many assessment -- sorry.

For example, decades of experience have shown that if we can capture the dominant physics to support MELCOR, if we can capture the geometry enrichment, moderated changes, and burn up. But we now have to do the assessment to see what are the dominant physics for each of these different non-LWR designs. And that's a work in progress.

So we'd have to look at residence times for pebble beds, when a pebble is next to a reflector, whether a pebble is next to a graphite blank, where it is inside the reactor core. We'd have to do all these parameter experiments researching these parameters.

Please note that there's a typo in my slides, and I included fuel salt in the SFR section.

It's not a salt, but we still care -- that is one of the parameters we're going to look at to change to see how the voiding in the coolant will affect what we care about in accident progression.

DR. PETTI: So you mean sodium, that's all?

MR. ALGAMA: Yeah, that was my fault.

MEMBER CORRADINI: Dave saw that.

MR. ALGAMA: So what is the current validation basis that we're leaning on with SCALE to move towards non-LWRs? SCALE has gone through a significant modernization effort in SCALE 6.2 and includes new features that are appropriate for non-LWRs. And this is TRITON MSR that you asked, Jose.

In SCALE, we have access to fast running production tools to verification capabilities if we use Shift. Further, we have access to multi-group data all the way through to continuous energy data that is appropriate for use for non-LWRs. To explore the different phenomena that SCALE treats, each of the phenomena in SCALE has been benchmarked.

So we have the decay heat benchmarks, destructive assay benchmarks, criticality benchmarks. And I'm going to hold on the criticality benchmarks a bit because that's how we use the criticality benchmarks to really measure the code performance and the curated nuclear data. And we use the destructive assay data in depletion. So with that depletion, with that nuclear data, we don't really know where we're going. There are many roads to Rome. And in SCALE, we validated our road to Rome, if that makes sense.

And I'm giving you an example of why it's important to benchmark. If you look at the cross sections for chlorine, between 7, 7.1. In this prototypic case, just that small change created a huge difference in K-effective. This is a prototypic analytical study.

MEMBER CORRADINI: Say that again slowly.

MR. ALGAMA: So if you look at the chlorine --

MEMBER CORRADINI: Slower generally. You don't have -- if they told you, you had to finish in 15 minutes, we'll give you 20.

MR. ALGAMA: All I'm trying to show you here is that small changes in nuclear data matter.

MEMBER CORRADINI: And the change between 7 and 7.1 is what?

MR. ALGAMA: Change the K-effective value by almost 2,000 pcm in this study. And if you look at neutron spectra, all these reactor designs are very different. So if you go back to my earlier slide, some operate thermal. So do thermal intermediate. Some do fast. We have to understand that this issue exists and how to translate that into our analysis tools.

MEMBER MARCH-LEUBA: And the root cause of that change was the chlorine-35?

MR. ALGAMA: Yes.

MEMBER KIRCHNER: And the cross section was wrong. It was -- that's what we use. You collapsed your cross sections inappropriately.

MR. ALGAMA: So our index data comes from Brookhaven. Brookhaven says, here are your --

MEMBER KIRCHNER: And they curate it?

MR. ALGAMA: They look at all the experiments, do their analysis, and say, this is good. So each library is good. It's up to the analyst to make sure how that library performs for that benchmark. So benchmarks drive what the data means.

MEMBER MARCH-LEUBA: So chlorine is not a very typical isotope to find in --

MR. ALGAMA: That's right. And if you look at that group, it's not really a commercial where the people in that group that helps finalize that data set.

MEMBER MARCH-LEUBA: They didn't want to do fine detail on chlorine. So I guess somebody made it with molten salt and said, oops.

MR. ALGAMA: Yeah, that's right. So it didn't show up till we did the molten salt problem. And we don't want it in graphite.

MEMBER KIRCHNER: Since you brought it up as an example. So they just chose to do a different energy averaging to get the different chlorine values in 7.1 or a different set of experiments to benchmark too. So what was the root cause?

MR. ALGAMA: So what was the cause between 7, 7.1? For that, I'm going to lean on the nuclear data expert that we have in the audience. Is that okay if I ask Brad to come and speak?

MEMBER CORRADINI: Yeah, but he has to sit in the chairman's chair.

MR. ALGAMA: So this was only intended to show that this matters and shouldn't be ignored.

MR. REARDEN: Brad Rearden, Oak Ridge National Laboratory. Yes, so I've been manager of SCALE for the past nine years and then passed it off to Will Wieselquist. He's here now. And so this particular chlorine cross section, there's actually no measured data what the cross section should be in this energy range. Nobody knows what the cross section is. It has not been physically discovered or explored yet.

So there are theoretical models of the nucleus, and they try to predict what the interaction of a neutron at that energy range could be. And so there was one prediction that came up and was distributed at 7.0. A different physicist gave a different prediction at 7.1. They thought it was better, but it makes a 2,000 pcm change --

(Simultaneous speaking.)

MEMBER CORRADINI: So you don't send it off to RPI to actually do the experiments in their accelerator?

MR. REARDEN: We're waiting to have somebody fund that activity. We've had several proposals come in. And so far, nobody has stepped up to actually provide the funding to get that done.

MEMBER CORRADINI: All right. Thank you.

MEMBER KIRCHNER: So have you done a survey of these non-LWR reactors and looked for vulnerabilities in the files?

MR. ALGAMA: So that's actually one of my other slides. And Brad actually has done the initial surveys and produced documents to start off that work. And NRC has decided to pick up on that effort. So kind of speaking ahead but yes.

MEMBER KIRCHNER: I'm just checking. Okay. Thank you.

MR. ALGAMA: Yeah. So benchmarks are important. So how are we moving towards non-LWRs? It'll be two tracks that we're going to try to follow. One is the fundamental data that we're going to use and then one is how we're going to apply it. So depletion is key here. Without depletion, we cannot support any of these codes moving forward. And if we can't deplete correctly, it's useless. So this is critically important and we have a large foundation basis in SCALE to show that we have a pretty good sound basis on depletion.

So data, to understand the validation, we need to assess the nuclear data for reactor class using system-specific composition and geometry spectrum, define accident scenarios and acceptable error, and look for appropriate isotopic validation cases to support this.

But in this area, we already know, as noted in slide 5 and 6, again, a lot of these reactors are over five weight percent. To support this use of over five weight percent, we need critical experiments, destructive assay data on high enriched fuel up to high burn ups which is used to validate completion methods.

For this in the past, there is some data out there that's beyond five percent and that's SNL's Neary work that was at seven percent. But beyond that, we don't really have much criticality data to support benchmarking it by codes. So when we do our assessment work, it's going to probably be highlighted for DOE to pick up.

MEMBER MARCH-LEUBA: So even when you're not with your miracle processing of theoretical chlorine-35 model, we do recommend the reactor designers to have additional margin just in case.

MR. ALGAMA: So that will depend on what the design is and how we react to it. So if they come in, like we're always saying, with huge margins, then the NRO will have to come and explain why that's okay --

MEMBER MARCH-LEUBA: And they also to have to have huge set on margins.

MR. ALGAMA: Yes.

MEMBER MARCH-LEUBA: Not be down to the last 900 pcm.

MR. ALGAMA: Right. And now we're hitting on the covariance data here which explains the uncertainty in the cross sections. So you can't really go beyond your measurement data.

MEMBER MARCH-LEUBA: Let's just put a bug in the stuff. When you review this reactor concept stuff, they need to be able to shut them down. So --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: -- there's a good example. I don't know where it's coming from, but this is a good example of how stuff reacted back in before my day which is back in Brad's day with a particular foreign reactor design we reviewed. And the staff had developed a tool to understand how the covariance data affected that reactor. And we found interesting discussion points with that design. So I hope that --

MEMBER MARCH-LEUBA: With a soluble light water reactor concept, you have the soluble boron that you can always play with.

MR. ALGAMA: Yes.

MEMBER MARCH-LEUBA: But with some of these reactors, you don't have all the regulatory rules.

MR. ALGAMA: Right.

MEMBER MARCH-LEUBA: So you will be right.

MR. ALGAMA: So with our process right

now, we have nuclear data benchmarked. We have SCALE benchmarked. It's initializing MELCOR and MACCS.

MEMBER MARCH-LEUBA: We're counting on you.

MEMBER REMPE: So I think I'm thinking about this. And to me, the advanced reactor program has been going on at DOE for several years --

MR. ALGAMA: Yes.

MEMBER REMPE: -- right? And now suddenly because you're involved, you've identified, hey, you guys have huge data gaps. And you've identified and said, so why not and DOE should pick it up. And of course they should or the design developer should've done it or somebody should've done it. It seems interesting that it had to come to you guys before somebody said, hey, you need data to go forward with these reactors.

MR. ALGAMA: I can't speak to that. All I will tell you is now that we're involved with supporting MELCOR, we have a particular process. We're following the same process that we would follow for LWR.

MEMBER CORRADINI: So along that line, I seem to remember with a Sandia report for Surry uncertainty for SOARCA and Peach Bottom uncertainty for SOARCA, the number one uncertainty was the decay heat.

MR. ALGAMA: Right.

MEMBER CORRADINI: So am I missing something here? That means that you're going to have to -- there's going to have to be a dedicated program because all of this fission, all of these --

MR. ALGAMA: Yes. So in my last slide, I said the decay heat is the most important thing to -- we need decay heat. We need destructive assay data, and we need -- I'm sorry.

MR. ESMAILI: Sorry. Did you mean --

(Simultaneous speaking.)

MEMBER CORRADINI: Well, the inventory that builds up drives decay heat which means that everything in essence is coming from this uncertainty that you're identifying or this group of uncertainties. So this isn't fair to ask you, but I could ask the DOE. But maybe they'll come up to the chairman's chair and answer it. So is there a part of the research program for these codes to essentially determine this in terms of these various -- or is it left to the applicant? I think that's what Joy is asking.

MEMBER REMPE: Well, just I'm surprised

that somebody -- I mean, when you think about long lead time activities that should be --

MEMBER CORRADINI: This is pretty long. This is long.

MEMBER REMPE: -- identified and underway, I'm also thinking of other design developers who are no longer there who had a reactor that wouldn't go critical. And I just am thinking about that maybe this should've --

MR. ALGAMA: So --

MEMBER REMPE: -- been thought of earlier.

MR. ALGAMA: -- experiments take a while to do. So I'm not sure that our work will say we need this experiment. I'm just saying that it's a possibility. So in the process of our assessment work, we're going to define what the issues are, are there validation data to support it? If not, is there another way of getting around it? And if we can't, then DOE or the applicant, whoever, please supply this information.

MEMBER CORRADINI: But I mean, another way to handle it is, okay, I have the inventory which I think is there given the reactor is operating in a certain fashion for so many days or months. And now I have a multiplier. Instead of these standard of 20 percent, I have a standard of one and 30 percent because I have uncertainties enough that I'm going to drive that.

MR. ALGAMA: So how do MELCOR and MACCS really define our margins?

CHAIR BLEY: Don?

MR. ALGAMA: Yes.

CHAIR BLEY: I'm trying to look through Volume 3 in your report and see some of the kind of things you were just talking about.

MR. ALGAMA: It's not there.

CHAIR BLEY: And it's not there.

MR. ALGAMA: No, it's not.

CHAIR BLEY: In molten salt you talk about maturity. You talk about some things. But I don't see anything about --

(Simultaneous speaking.)

MR. ALGAMA: Correct. And because this is just a plan to get us started. So in this LWR slide, I'm trying to say, oh, you've already seen what our plans and I'm trying to explain what are our unknowns we're going to try and hit with this assessment work.

MEMBER KIRCHNER: So in this section of the report, we don't have to really see what the key gaps in the data.

MR. ALGAMA: Yes, yes. There could be more. There could be less. We may be able to mitigate it with analytical methods. It's a research project that just started.

MEMBER KIRCHNER: Let me probe a little more. Pick on the -- I had it on. That bullet there, the ENDF/B-VIII graphite data, I would've thought graphite was something we knew fairly well, at least Great Lakes carbon graphite. So --

DR. PETTI: I used to believe that too.

MEMBER KIRCHNER: So the issue here is --

DR. PETTI: I used to believe that, but I've been disabused by this.

DR. PETTI: Okay. Well, throughout the nuclear industry's history, we've used a lot of graphite. We have done a lot of experiments that involved graphite. I think in many cases, we were able to predict whatever the objective of the experiment was, criticality, whatever. So what's the problem here and how big is it?

MR. ALGAMA: So the problem here is that the new data is simulation-based and not experimental-based. So we'll have to compare the new data in ENDF-VIII to actual reactor grade graphite and see what the differences are. I'll give you an example.

For HTGR and criticality benchmark, there's a half millibar change in the carbon capture cross section from ENDF 7 to ENDF 7.1 which caused a 1,000 pcm change in K-effective. So --

MEMBER KIRCHNER: Run that by me again.

MR. ALGAMA: So there was a half --

MEMBER CORRADINI: Slower. I want to absorb what you're saying.

MR. ALGAMA: So between -- all libraries are correct. It's up to the analyst to show how. So --

MEMBER KIRCHNER: Trust but verify --

MR. ALGAMA: Trust but verify.

MEMBER KIRCHNER: -- is what you're saying, yeah. I get that part.

MR. ALGAMA: So Brad makes -- the data, Will shows why it's right. So between ENDF 7 and ENDF 7.1, there's a half millibar change in the graphite capture cross section which -- so when you --

MEMBER KIRCHNER: In the thermal range.

MR. ALGAMA: In the thermal range. And when you move it out of the K-effective calculation, that little blip resulted in 1,000 pcm K-effective change in the result. So these are things --

DR. PETTI: There is so much graphite.

MR. ALGAMA: There is so much graphite, exactly.

DR. PETTI: So you're looking at the cross section change, that's tiny, right? But then there's so much graphite. That plus the impurities --

MEMBER KIRCHNER: Well, that's where I was going to come in, right?

DR. PETTI: -- we're just talking about.

MEMBER KIRCHNER: Are you talking about impurities or --

DR. PETTI: I think it's a big deal. It's probably a combination. In the old days, they don't know the impurities well enough.

MEMBER KIRCHNER: Right.

DR. PETTI: So you're going back to old benchmarks where you're just inherently limited. We can certainly measure more stuff now in S alpha-beta. But we can't go back and figure out what the impurities were. The better thing is today's graphite is so much cleaner that that stuff. I wouldn't get all upset that you missed the benchmark because we get asked this question all the time. Today's graphite is not yesterday's graphite.

MEMBER MARCH-LEUBA: The problem is after 60 years of working with thermal and light water, we're so used to seeing 1,000 pcm is a terrible level on thermal and light water. But in this unknown component, it might not be and it'd be really, really embarrassing if you have a ten billion dollar facility and it doesn't fire up.

MEMBER KIRCHNER: Okay. But just so I understand. This difference in the VIII library versus the VII which just the model that they put in, in the thermal spectrum.

MR. ALGAMA: Yes, for the cross sections.

MEMBER KIRCHNER: For the cross sections.

MR. ALGAMA: So they made a physics model and used that in lieu of an experiment.

MEMBER CORRADINI: You give us great confidence. Move on.

MR. ALGAMA: We've been doing it that way for 60 years.

MEMBER CORRADINI: Yeah, but you have a lot of LWRs working, so you can do benchmarking.

MR. ALGAMA: That's true.

MEMBER CORRADINI: That's actually huge. Yeah, that's big.

MR. ALGAMA: So there's some --

MEMBER KIRCHNER: That puts a lot of uncertainty into --

MR. ALGAMA: Correct. We'd have to quantify that somehow.

MEMBER KIRCHNER: Have to quantify it, and that has implications. For how much -- well, this a methods, not a design presentation. But it raises a lot of serious questions for a design review.

CHAIR BLEY: WELL, about design and even operations. Because if you have uncertainty, the way you're going to start up could make a big difference. You probably need a neutron source in there to get lots of subcritical multiplication to see what's really going on before you actually try to take it critical.

MR. ALGAMA: So just the last point I wanted to say is for non-water cooled thermal reactors. We need S alpha-beta data which we don't have. So for FLiBe and FLiNaK, the data doesn't exist. So that's another item that we're probably going to say, this is important. DOE or someone, please develop it.

So we've talked about fundamental data. Now we're going to talk about how we're going to validate our methods tools to show that it works. So we're building off, again, the vast validation basis of SCALE, the 800 criticals, hundreds of PWR-BWR benchmark destructive assay data, the decay heat data that we benchmark to. And we're going to have to get data or find data to support the extension.

So we're talking about how are we going to support depletion analysis, isotopics and decay heat results with burn ups matching these specific designs, fission productions in salts, metals, TRISOs, et cetera. All are expected to be needed. We have additional gaps in validation for criticality at high temperatures that use reactor grade graphite and with salt coolant.

The figure below is one example of data from a foreign radiation facility that's well suited for validating SCALE because it also includes cesium isotopes. So there's a lot of data out there. Not all of it's benchmark quality. So we have to review and assess and see what the data is, how that data can be used, and which part of our codes does that data hit for benchmark.

CHAIR BLEY: Let me ask you a question about -- because some of this jumps off the page actually. These are pretty big deals. It almost seems like it would be appropriate to author some kind of a technical paper so potential designers of such systems get a heads up from the NRC about where some of these big gaps are and what they might mean to their designs.

MR. ALGAMA: I agree, and I think the starting point is the nuclear data performance assessment for advanced reactors report and the ENDF-BVIII report for covariance data.

CHAIR BLEY: Are these all in there?

MR. ALGAMA: These are all --

CHAIR BLEY: The ones you talked about?

MR. ALGAMA: Yeah, yes. Now the things for --

CHAIR BLEY: I haven't read those.

MR. ALGAMA: So some things like the S alpha-beta are not in there because this is specifically for nuclear data. The things that are missing for S alpha-beta, correct if I'm wrong, Brad, will come in our assessment report. As we're using this, what are we missing? Is that correct? Brad is nodding his head that I'm not giving you --

CHAIR BLEY: When do you expect a report like that to --

MR. ALGAMA: We just started.

CHAIR BLEY: -- be put together? I understand.

MR. ALGAMA: Maybe in a year. Is a year

a good time, Will? He's shrugging.

MEMBER CORRADINI: Don't make them agree in the middle of this. But I think you're saying that this is an update in Volume 3 that we have here. This is the living document this would appear in?

MR. ALGAMA: Yes, yeah.

MEMBER CORRADINI: Okay.

MR. ALGAMA: We would have a report that shows how we've assessed our codes for LWRs and this will be all included. And any data gaps that we --

CHAIR BLEY: So that will come back into this?

MR. ALGAMA: Yeah. It'll actually be a separate report.

MEMBER CORRADINI: But referenced?

MR. ALGAMA: Referenced.

CHAIR BLEY: That's what I thought you were saying.

MR. ALGAMA: Yes. So what's the current focus? So how will data transfer really work for all these designs between MELCOR, MACCS, and SCALE? That's something that we're going to have to -- we're currently dealing with. What's the most efficient way? What's the simplest way? We don't want to be in a position where we're generating cross sections and depletion data. We want to be in a position where we have a reactor that we can just pull from. Those analyses takes a few second to run.

MEMBER MARCH-LEUBA: With this reactor we're talking of here this afternoon that expand, how do you do this calculation?

MR. ALGAMA: So we're doing bounding analyses. So from --

MEMBER MARCH-LEUBA: Yeah, but eventually you'd have to have a lot of this.

MR. ALGAMA: We can -- yeah, so we can -- the first approach is, can we bound the expansion? The licensee will come to us conceivably and say, our reactor, as far as a design, move this way couple of millimeters. For the types of needs that MELCOR and MACCS have, that's not really going to change our --

MEMBER MARCH-LEUBA: I was thinking MELCOR --

MR. ALGAMA: -- isotopic information.

MEMBER MARCH-LEUBA: -- was thinking of this.

MR. ALGAMA: Yeah, I'm not there.

MEMBER MARCH-LEUBA: Well, you're scared. You provided cross sections for them too. And if we are relying on that expansion for feedback --

MR. ALGAMA: We can do it. We do it. We use Shift and COMSOL and some other codes, correct if I'm wrong, Brad, that we had a feedback effect of data cross sections.

MEMBER CORRADINI: But you do it -- but you do it -- you don't do it -- you do it in a manner where you're basically taking it between one at -- kind of what Jose was saying before. There was some sort of time scale where you look at quasi-steady slices of it.

MR. ALGAMA: Yes.

MEMBER CORRADINI: And then look at the effect over the slices.

MR. ALGAMA: We iterate back.

MEMBER MARCH-LEUBA: You have to apply it ahead. You cannot --

MR. ALGAMA: Correct.

MEMBER MARCH-LEUBA: -- wait until it's done.

MR. ALGAMA: Yes, and that's why this assessment is so important. We're trying to do as much of this ahead as we can that's reasonable.

MEMBER KIRCHNER: Can we go back one slide? Will you go back one slide, please. That first bullet, what's the general implication for generating a mechanistic source term?

MR. ALGAMA: We don't generate mechanistic source terms.

MEMBER KIRCHNER: I know you don't --

MR. ALGAMA: So --

MEMBER KIRCHNER: -- but you give the initial conditions.

MR. ALGAMA: Yes, so we're saying for depletion, this is the data -- the kinds of data we'll need to --

MEMBER KIRCHNER: So I repeat my question. What's the implication in the endgame?

MEMBER MARCH-LEUBA: Let me -- in K-effective, 1,000 pcm is allowed. That's one percent of the K. One percent in your source term, you'd be lucky to have it. I mean, you'd be lucky to get part of it too.

MR. ALGAMA: Yeah, we have to translate that and send it down, and sometimes it's not meaningful. So it looks -- I'm sorry. So reviewing the LWRs, you need some calculations. We have a higher acceptance criteria for a 2D lattice code. And then usually by the time you get down to the calculations that matter, that uncertainty is gone. It's disappeared.

MR. ESMAILI: These uncertainties are propagated. So as I showed you, there are so many processes --

MEMBER KIRCHNER: There are. That's why I'm asking.

MR. ESMAILI: -- where you find in the system, in the containment, and boundary conditions. This all affect the source term. So these are, like, overwhelming and then you go further. So Jon is going to talk about it. You go further down, then you have uncertainties on the dose, right? I mean, how the source term gets translated into an atmospheric dispersion. So what he's talking about is probably not that simple.

MR. ALGAMA: Yeah, we don't need to be as -- we want to be as simple as possible. We don't need to sharpen the analysis to the point where it's meaningless for this analysis process. We want it to be fast.

So some of the current focus, moving on. How are these codes going to connect for each of these design problems? How are we are we going to deal with moving fuel with MELCOR and MACCS -- sorry, MELCOR and SCALE? What is the level of chemistry that should be captured in SCALE versus MELCOR? This is some ongoing topics.

And in summary, it's pretty much like what Jose said, leverage existing, codes, capabilities extended where it makes sense. Hopefully define some experimental needs to help us with assessing the assumptions in physics that we put in are correct. Enhance what we already have and then plan will be updated as we learn more. So that's all I have.

DR. PETTI: So I'm just a little bothered that this sounds like the sky is falling.

MEMBER CORRADINI: That is what I --

DR. PETTI: I hope we're not taking that away. I mean --

MR. ALGAMA: No, that isn't my takeaway.

DR. PETTI: Okay. Because there will be sodium reactors built and many gas reactors built. And no one said, oh, we couldn't overcome the uncertainty in the cross section. We're still able to start and we're still able to operate.

MR. ALGAMA: That's what the understanding is.

DR. PETTI: Right. I don't know. I mean, I haven't got enough salt to know whether it would actually affect those sorts of things. But these are uncertainties that the designers at least I think are aware of and have compensating measures to deal with.

MEMBER CORRADINI: But we listened to reactor physicists and they get nervous over pcm.

DR. PETTI: I guess so.

CHAIR BLEY: And we're not sure of all potential design curves. We're pretty sure of some.

MR. ALGAMA: So the main areas here is we're really hitting on depletion and criticality methods with benchmarks. We can't extrapolate, but we would like to have benchmarks to extrapolate.

DR. PETTI: So I know many new stuff for the TRISO. We've got a lot of data coming in. Some fission products, we hit on the head. And others, it has to do with the yield curve. And fission products I mostly don't worry about but we can measure. We do see deviations and we assume because we don't know the yield well enough which is sort of interesting. I don't think it'll make a huge difference in the final analysis, but you'd think we knew the yield curve really well because we've been fissioning for a long time. But we really don't.

MR. BARR: So how are we with time?

(Simultaneous speaking.)

CHAIR BLEY: Take whatever you need. We're the last ones on today.

MR. BARR: Okay. My name is Jon Barr, and I'm currently the Acting Branch Chief in the Accident Analysis Branch in Research. We have responsibility for the MACCS code suite. And in this presentation, I'm going to be talking about the MACCS code, what it's used for, and different regulatory applications and then why it's suitable for non-light water reactor consequence analysis and then some of the plans that we have for potential changes, specifically for non-light water reactors.

As Don and Jose have said before, this is a collaborative effort. We have been supported by Sandia National Labs, particular Dr. Nathan Bixler. On my branch, Dr. Keith Compton has also contributed to some of the thinking here.

And I just want to start by an overall message that relative to some of the other codes that we've heard about, MACCS is much more technology neutral. And many of the examples of potential code development that I have in the report and that are discussed and I'll discuss now are areas where we at least want to do some further evaluation to identify whether or not some code development is needed.

MACCS is the only code used in the U.S. for probabilistic offsite consequence analysis. There is no DOE or industry code that performs this specific function. It is used by NRC, by Department of Energy, by nuclear power applicants and academia. It's also used internationally and shared alongside MELCOR and CSARP, Cooperative Severe Accident Research Program. With about 25 or so different member countries, I think we have on the order of about six or seven hundred different MACCS users worldwide.

MACCS is very flexible and it can treat all of the different technical elements that are in a Level 3 PRA standard including radionuclide release, atmospheric transport, meteorology, protective actions, site data, dosimetry, health effects, economic factors, and uncertainty.

On the bottom left is just a little schematic showing some of the processes that are treated in MACCS and some of the exposure pathways. We're seeing just some radionuclide release into the atmosphere, some transport wet deposition in this case with some rainfall, and then some exposure to people either through direct exposure, ground shine, or even through animal pathways.

CHAIR BLEY: Jon?

MR. BARR: Yes.

CHAIR BLEY: Since you showed a nice

picture with a stream running through it, does it also track the movement through the stream?

MR. BARR: MACCS treats the results of --

CHAIR BLEY: It doesn't do groundwater I'm pretty sure, right?

MR. BARR: Correct.

CHAIR BLEY: Yeah.

MR. BARR: But it treats the results of deposition onto water bodies. But it does not treat releases in a groundwater aqueous form.

CHAIR BLEY: But does it track them in a water body?

MR. BARR: The result can be --

CHAIR BLEY: Yeah, there might be --

MR. BARR: -- ingestion --

CHAIR BLEY: Yeah.

MR. BARR: -- from drinking water.

CHAIR BLEY: And it does track the water streams, say, to a city's water supply or something like that downstream. I didn't know it did that.

MR. BARR: It probably is not going to go into that level of detail.

CHAIR BLEY: So it'll tell you what went in the river but not where it went.

MR. BARR: Right. So MACCS will have

essentially for each of the elements in the polar grid will have a land fraction and a water fraction. And it will consider that, some of the deposition onto the water given the water fraction and the land area. And then it will compute a dose.

MEMBER CORRADINI: So it doesn't connect it from grid to the next. That's what you're saying pertaining water. Okay.

MR. BARR: Correct, correct.

CHAIR BLEY: There are other codes that will do that, I know.

MR. BARR: Yes, correct. There are other codes that do that.

CHAIR BLEY: Does NRC have them, or is that just something -- I know DOE does.

MR. BARR: We have other codes that are used for groundwater.

CHAIR BLEY: Okay.

DR. PETTI: And so in terms of tritium, from what understand, it's very important. The limit in water is extremely low. You may need to rethink or use one of these other tools to convince yourself.

MR. BARR: That's right. We may need to augment this capability and there are other tools that have similar -- an overlap in some of the models that we can use here.

Quickly I want to show an animation that demonstrates the Gaussian plume segment model within MACCS. It's a relatively simple model. But when we have hourly plume segments with different weather data corresponding to each hour, it can develop a somewhat complex deposition pattern. And MACCS is used in a probabilistic fashion, so this is showing one individual weather trial. But in a typical MACCS calculation, we would run this about 1,000 times and look at the statistics over all of the different plots.

CHAIR BLEY: I haven't thought about this at all. It just comes off the top of my head. In chemical plant risk assessments, you have a lot of times where you have to have your air mixtures. And I don't think MACCS handles that sort of thing, the first part. The second part is, are there any heavier than air releases that we could expect out of some of these interesting design we might use?

MR. BARR: That's a good question. And --

CHAIR BLEY: Thank you.

MR. BARR: -- I'll talk a little more about that in a few more slides.

MEMBER CORRADINI: The original MACCS is

2D, but I thought you guys were making some changes, if memory serves me.

MR. BARR: Yes. You're referring to the fact that we are in the process and towards completion of integrating 3D atmospheric transport dispersion model.

MEMBER CORRADINI: That's HYSPLIT, isn't it?

MR. BARR: That's HYSPLIT and that can consider the temporal and spatial variation of the wind field. Whereas in this model, it's simpler in that a plume segment will go in the direction that the wind is blowing at the time of its release. There is no curving of plume segments.

MEMBER KIRCHNER: So when you do that, Jon, is that -- I'm trying to think through this. Is that conservative when you apply these weather trials to dispersing the source term, the release, where I'm going with, versus having a 3D model? I mean, do you see where I'm going with this? Is it in the 2D application with these different weather trials and the source term and its evolution and all the rest? Is that generally conservative, vis-a-vis putting the HYSPLIT in?

MR. BARR: So I'll get into -- so the

issue that's more --

MEMBER KIRCHNER: Except for weight defects and all of that.

MR. BARR: -- relevant in terms of the near-field issue. And so the HYSPLIT integration is not really going to be as beneficial in that near-field domain.

CHAIR BLEY: There's something -- and you guys can correct me on this. This was going on 20 years ago. There were people who developed models that don't just assume the constant direction but allow the wind to change and move it around. And to the consternation of the folks who developed those models, in all the benchmarks, it seemed not make much difference unless you're tracking for evacuation and that sort of thing.

MR. BARR: There have been some benchmark studies that have compared the Gaussian plume segment model to more advanced models. And when you look at what the averaging results over all the different weather trials, they compare pretty well in that averaging kind of format. Not as well if you have one specific weather trial you're running.

MACCS can also calculate a lot of different consequence measures including dose but also health effects in terms of cancer fatality risk or cancer incidence risk, the extent of land that's contaminated, the population that can be displaced by the protective actions, and the economic costs that can result from an accident.

There are several different regulatory applications of MACCS. We use it for regulatory cost benefit analysis. Some examples of that kind of analysis are ones that were performed in the containment protection release reduction study and also the expedited spent fuel transfer study in which we looked at different alternatives and used MACCS to calculate the offsite consequences to inform those decisions.

Applicants for license renewal typically use MACCS in the environmental report for severe accident mitigation alternative analyses. And in the design phase, similar calculations, severe accident mitigation design alternatives, or SAMDAs.

We use MACCS in Level 3 PRA studies as is currently being done in the full scope Level 3 PRA project. It's been used extensively in different research studies of consequences, notably the SOARCA studies going back over the last decade. And it's also used in support of emergency preparedness and planning. MACCS can compute a probabilistic dose versus distance, a calculation to inform the likelihood of exceeding a dose at a given distance to inform emergency planning zone size decisions.

MEMBER REMPE: So it's also in the DOE toolkit, right?

MR. BARR: Correct.

MEMBER REMPE: And so it actually goes just beyond the regulator is my understanding.

MR. BARR: That's right. We're actually updating the version of MACCS that is in the DOE toolkit.

CHAIR BLEY: Is it possible -- I know there are other codes that do this. Is it possible to use it in a -- and not looking at all these weather patterns but in a realtime application of weather at a certain location where a release is going on?

MR. BARR: If we had --

CHAIR BLEY: Or does it lock in to the wind direction and stay the same?

MR. BARR: -- the data, we could -- if we had the hourly data or a forecast for all of the parameters, wind direction, wind speed, precipitation.

CHAIR BLEY: Or not the forecast but actually what's going on, could it?

MR. BARR: It could, but we have also RASCAL which is better for that application.

CHAIR BLEY: Work better for that. Yeah, I see.

MEMBER CORRADINI: Why does it work better? I can't remember.

MR. BARR: RASCAL?

MEMBER CORRADINI: Yeah, similar?

MR. BARR: Similar --

(Simultaneous speaking.)

MEMBER CORRADINI: It's a puff model like MACCS, isn't it?

MR. BARR: It has puff and plume, a plume segment and puff model. But RASCAL has a way -- has a process for integrating realtime weather data.

MEMBER CORRADINI: Right. It has an input that is more friendly to do it if you knew it.

MR. BARR: Okay. Let me get through this slide first. Okay. There are other codes that NRC develops that have some overlap, particularly with respect to atmospheric transport and dose calculation capabilities. MACCS is really best suited for consequence analysis. It's very flexible and almost all of the numbers within the code are user inputs that a user can change and adapt for a different application. It's been used for lots of different types of sources including reactors, spent fuel pools, and dry cask storage releases. And as I said before, it is generally a very technology-neutral code.

Two of the primary alternatives that exist with some similarity are RASCAL and RADTRAD. These will be discussed in Volume 4 of this report series that I imagine you'll see at some point in the future. Both of these are dose calculation tools.

RASCAL is used in incident response to inform protective action recommendations. It does not consider the full range of protective actions and consequence measures.

RADTRAD is used for design basis accident calculations for compliance with NRC siting criteria and for control room habitability.

MEMBER REMPE: So in our research review a couple years ago recognizing that code minimization is beneficial for a cost perspective because you've got to maintain all these codes, upgrade the documentation, et cetera, we suggested trying to find a way to have one code that integrates these different capabilities. And these have a switch they turn off and make MACCS more like a RASCAL evaluation. And has that gone anywhere? I know Mike came to talk to us a few weeks ago and said that you were thinking about it. But we didn't get into the details of what you were thinking about. What's the status of doing that?

MR. CASE: We haven't lost that thought yet. So once again, it's the volume you can see. So there's a rad protection volume that we're intending to rate for non-LWRs. And so we're trying to orient it towards consolidating the codes.

MEMBER REMPE: Yeah, I think that's an important point.

MR. CASE: We're not smart enough to see whether I can pick it up in MACCS. So I want MACCS to continue on its development path and not really hold it back. But sort of the implication in your recommendation that we talk about is can we stuff all these things into MACCS. I just don't have the flexibility to do that yet. But we haven't lost that idea either.

MEMBER REMPE: Thanks.

MR. BARR: My thought is that RASCAL and RADTRAD are in active development-wide user basis. They're probably two that what might make some sense to keep separate. But there are several others that are not as actively developed and used that may make sense for some kind of integration with other codes.

So non-light water reactor consequence analysis, we've gone through a process of trying to think of all the different ways that MACCS as it is currently coded in Version 3.11 would be challenged for handling a non-light water reactor accident release and consequence calculation. And primarily the issues that are specific to a design are related to the radionuclide particles that are released.

The different properties can affect the atmospheric transport, the deposition, and then different -- the biological interaction essentially through the dose conversion factors.

CHAIR BLEY: Are the dose conversion factors and the uncertainty in them well known for radionuclides other than the ones we're used to seeing in light water reactor accidents? Is this a gap for you guys that is something you need to worry about?

MR. BARR: So one of the issues is regarding the radionuclide particle size distribution. And so the dose conversion factors that we typically use assume a median particle diameter of one micron. So if the particles are going to be significantly smaller than that, we may want to evaluate whether or not we need to use different dose conversion factors or modify them somehow.

CHAIR BLEY: I think that question might be involved in the things you already have up there.

MR. BARR: I think so.

CHAIR BLEY: Okay. I'm not sure.

MR. BARR: Regarding radionuclide screening, MACCS includes a large library of the different radionuclides. But in a typical analysis, you only use a subset of that for a consequence analysis. So we have a set of 69 radionuclides that are typically used for light water reactors. We would need to likely augment that with potentially some activation produce nuclides that we might see.

I mentioned before the difference in potentially smaller particles that could be released. We might see some different chemical forms of radionuclides. In a sodium reactor, if iodine is released as a sodium iodide chemical form instead of cesium iodide, that could impact the chemical form of the cesium that's then released. And the chemical form can impact the dose conversion factors.

And then also MACCS has some places in it where it will assume a spherical particle in terms of the calculation for dry deposition. If we have significantly non-spherical particles that are released, that may be something that we may need to correct for in the dry deposition velocity formulas.

And tritium is an issue that is also something that will be evaluated for other codes like RASCAL and RADTRAD. And that may warrant some changes to the dose conversion factors. We may need to also add a different chemical group within MACCS to treat tritium separately.

DR. PETTI: Is it -- tritium is not in MACCS today?

MR. BARR: Tritium, I believe it is in MACCS. Whether or not it's released, we have the mechanisms to provide for the transport and consequences. Tritium is not typically in the source term. That would come from MELCOR.

MEMBER CORRADINI: Because it's too small compared to everything else? It's there. It's just too small, I assume.

MR. ESMAILI: Yeah, this is another concern, but --

DR. PETTI: Internally fission.

MR. ESMAILI: But that's what I said. For MSRs and FHRs are going to be a molten salt. We are going to add tritium as an additional class and we are going to track it. And that information is going to be passed onto us.

DR. PETTI: It's going to be a big difference whether it's HT or HTO, the conversion in the environment. I had mentioned to Kimberly at break the state-of-the-art tool is a German tool called UFOTRI, U-F-O-T-R-I. I googled it. It's in IEA. It seems to have accepted this is -- mine is ten years old, a fission program. But it did all that conversion in the environment because it's very different than other materials because it's hydrogen. So there's a lot. There's data. They did dispersion tests in Canada. So there's lot of data to benchmark.

MR. BARR: And then the area of code development that we've made the most progress in so far is related to near-field atmospheric transport. We prioritized that since it's really a site related issue that can apply to all the different non-light water reactor types. And so I'll just briefly kind of discuss our thinking on that.

MEMBER REMPE: Just a question real quickly. With all the fusion reactor analyses that were done through DOE, did they not ever use MACCS and consider trading?

DR. PETTI: We did use MACCS. But for the tritium work, we deferred to the German model, this UFOTRI, because it was considered to have much more detail in it.

MEMBER REMPE: But there might be some insights that --

(Simultaneous speaking.)

DR. PETTI: Right. There's a lot of stuff that was published. Like, you've got to go before Google to find it.

CHAIR BLEY: Jon, just to get ahead of you. Are you trying to build near-field transport into MACCS, or is it something you'll do separately? Because it's different.

MR. BARR: We have -- so yes. So kind of the point I want to make in the next two slides is that we are in the process of reviewing different modifications to the existing Gaussian plume segment model that do a better treatment of building like events. So just to illustrate that the --

MEMBER CORRADINI: Isn't that the HYSPLIT ability or I thought that was. That's not HYSPLIT's ability? That's --

MR. BARR: HYSPLIT is more on a broader spatial scale.

MEMBER CORRADINI: So spatial scale is not appropriate for this?

MR. BARR: Right.

MEMBER CORRADINI: For HYSPLIT?

MR. BARR: This is not a HYSPLIT. Yeah, this is -- so the MACCS user guide essentially cautions not to use the existing Gaussian plume segment model closer than about 500 meters. This is just a simple schematic showing that within a certain distance of the buildings, you can have these recirculation cavities that can increase doses or concentrations there.

And so there are a lot of different options that treat these issues. And there are simple ones that are used in NRC codes like ARCON96 and PAVAN that have modifications to Gaussian plume segment models. And then there are also some complex ones that are shown here. This is an example by Los Alamos called the QUIC model. And it can do a fairly detailed representation of building SCALE dispersion. And this could be used --

CHAIR BLEY: Can you couple the results of this analysis to the downstream for typical MACCS analysis? Or will you being using this separately just to look at onsite, near site?

MR. BARR: Assuming we proceed with a modification to the Gaussian, it would be all part of the same model. This slide was just to illustrate that there are lots of different types at different levels of complexity for this treatment. However, when we consider which makes sense in this, the application, one of the most important -- some of the important ones are the cost and time efficiency, the resources, and essentially feasibility for probabilistic application.

CHAIR BLEY: Yeah, just I'm thinking back to what Jose talked about earlier on the other code. Given the fairly simple treatment of the downstream winds and that you don't let them meander as they do, putting this in the front end seems like it's complicating your problem beyond what's really going to make a difference.

Now if you want to know what's happening on site, this is crucial information if you're worrying about worker exposure, that kind of thing. But putting them all together seems ambitious or at least to me without much thought --

MR. BARR: I agree.

CHAIR BLEY: -- without much thought.

MR. BARR: Yes. Integrating something complex like the QUIC model would be a fairly major research project that's probably not going into this application.

DR. PETTI: But again, remember many of these events, the developers want the site boundary to be the fence. And that's not 500 meters. That's usually 400, 325 meters sort of number. And so --

MEMBER CORRADINI: But they would have to do something.

DR. PETTI: Right. And so --

CHAIR BLEY: But you wouldn't have to integrate these two --

DR. PETTI: With one tool and then --

CHAIR BLEY: One tool and the long distance.

DR. PETTI: But again, to meet those protective action guidelines at the fence, MACCS may not be acceptable as-is, right?

CHAIR BLEY: Oh, I think you're right.

MR. BARR: So our current status in this area is that we have reviewed lots of different options for improving building wake effect treatments in the near-field domain. And before we go ahead and integrate one of these modifications to the Gaussian plume segment model, not the fancy QUIC model that I showed but some of the more straightforward modifications.

Before we do that, we are going to do a

comparison of MACCS with different parameter values to be more -- as sort of a conservative style calculation to compare to these other codes so that we can better understand if someone wanted to use MACCS within closer than 500 meters, let's say down to 200 meters or something, how confident can we be that that MACCS is going to be conservative?

And so we can look at different weather. This would be using MACCS for the point source with no buoyancy and a ground level release. So we know that it would be conservative or we would expect that it would be conservative like this. But the goal of this sort of comparison is to be better informed about whether or not MACCS really is conservative and with what input parameters.

Based on this, we could then proceed with integrating one of the more simple Gaussian plume segment modifications. And then along with that would be some method of communication of guidance for a potential applicant in terms of what we might recommend as a way to parameterize MACCS with this model so that we can be sure that they're conservative about this close in distance. And if they can't tolerate that conservatism, then we would know that and they would have to have some other alternate approach.

One of the things that Jose mentioned as you concluded that this is a living document. And as we proceed and spend more time looking into this, our thinking on this may change. And we are attempting to identify whether or not we need to actually do the code development before we do it. So we're going to probably have to figure out whether or not these changes would really make a difference that's necessary. And that's it from me.

CHAIR BLEY: Thank you very much. I think now we want to go back to that original slide package. And Mike is going to give us a wrap up.

MR. REARDEN: Yes, I'm Mike Case. I'm substituting for Kim. Kim had to go. She had an appointment early on, so she had to leave. So I just saw her slides at noon, so I will --

(Laughter.)

MR. REARDEN: -- try my best on it.

CHAIR BLEY: Can I interrupt you before --

MR. REARDEN: Sure.

CHAIR BLEY: -- you go ahead? Just because I had a couple conversations, and it seems the staff has changed its mind at this point. Some of the things Mike's going to show us, they're actually going to want a letter. And I want the committee to think about this before we get around to going around the committee and think about the things Mike shows us.

It doesn't need to be this month. It could be maybe early next September. So sometime in there. But I'm going to ask all the members, especially after seeing what Mike summarizes here, what additional information we ought to ask for before we would hit that point. So just for you think about.

MEMBER CORRADINI: In terms of another subcommittee you mean?

CHAIR BLEY: Perhaps or a longer full committee than we would've thought.

MEMBER REMPE: And so interrupting, that's news because I did not know they were thinking about that.

CHAIR BLEY: That is news. It's news to me.

MEMBER REMPE: Yeah. Is the staff receptive to some changes in their approach or they definitely want these two out of the -- or three out of the five documents to be the basis of that letter? I mean, we can ask for additional information, but you're not planning to do anything differently, the approach the way you've outlined it in those documents.

For example, there's been several comments by several members saying two different documents from the same agency indicate that more crosstalk is needed. And you don't want to try and address -- again, if we remember comments that you're not receptive that you want to stay with the documents the way they are.

CHAIR BLEY: Right.

MR. CASE: Right. I would think more of them as like a working draft. This is an evolving area. And so really we'll go right to the second slide.

CHAIR BLEY: That's where the meat is.

MR. CASE: Yeah. On the first slide, the point is what we've been directed to do by the Congress of the United States, the Commission, and the advanced reactor folks is we need to be ready for these things. They are not fooling around. They don't want to hear that I got to wait for these details to come in. They want us to start being ready. So they are very serious that we got to be start being ready.

So we want your insight on the three plans that we have. And I would think in the past, sometimes you would have very long issues where you would start a letter and then add to the letter as things progressed. It's probably that model we need to think differently. So we can't sit here and say, I want to wait for all the details before I start to think about what you might suggest.

You heard a lot of very good ideas. I think the committee needs to get together, consolidate those ideas, and give me the top five of what I should start to worry about. And you're damn right. We're going to start to worry about whatever you say. This is a huge area, so we're going to worry.

We crafted some questions because people want us to think differently. So some of these are -- some of them are pretty -- the gaps, that's a pretty typical thing. Some of these are a little bit more of should we think differently as an agency. And so, like, number three, some people suggest, hey, why don't you guys just use the codes that walk in the door? And so we say, hey, we could do that. But I think that leaves a lot off the table.

So we want to get people's input on things like that. Why would I have to develop all these codes? I think they're very valuable because in the developing of the codes, we get an understanding of these designs. If we wait until we get the application and say, hey, here's a code, we're going to have to learn. There's all sorts of downsides to doing that. But that's one possible way to do it.

Some people think -- practically, they look at the time that we have to review, like, a small reactor. They say, hey, not a lot of time to do a confirmatory analysis. So why bother? Why don't you just do some back of the envelope estimation of these things? So those ideas are out there. So we need your thoughts on whether it is worthwhile to do all this code development.

MEMBER SKILLMAN: Mike, let me ask this question. In my experience, one of the most important things is staff doing the work. So the staff is exercising the cadence and the thinking process to know what they should come up with as an answer. In other words, it is the drill. It is the exercise. It is the intellectual and physical activity of doing the work that ensures the final product delivers.

I could interpret from what you just said an acceptance of we'll just use a shelf code. We're not going to verify it because it's been used elsewhere. And we can run with that because it's accepted practice. I don't think that that's what you meant, but I could interpret that.

MR. CASE: Right. All I'm saying is that one option that people could propose. Would I recommend that path? Probably not. I, 100 percent, agree with you. But there are people out there that say those types of things. So that's why we're seeking feedback, not only from the ACRS but when we put these plans out publically and we get reactions from industry and the other groups. So some of these things are a little bit provocative. But we want to at least consider thinking differently.

CHAIR BLEY: Now Mike, because this is new to us today, I had a talk with Kim about this when she brought it up. And I want to make sure we're not getting crossways between various people on the staff. But she indicated that you didn't need this tomorrow. You didn't need it -- maybe in a couple of months or even by September would be useful.

We had asked that this session focus on kind of high-level things and gaps which you did. And she pointed out to me that that leaves out a lot of details which is does and which we were willing to do, especially because we didn't think we were writing a letter at this time. So we might want to go more in depth on some of this. Is anything I mentioned crossways to what your expectations are?

MR. CASE: No.

CHAIR BLEY: Okay. So I think we'll talk about it as a group. But I'm thinking we ought to wait to even get serious about this until we get the transcript back and everybody can read it. Any of the members who have thoughts on these, somehow today but also over the next month, feel free to send them to me. And then we'll figure out --

(Simultaneous speaking.)

DR. PETTI: Mike, I think we'll want --

CHAIR BLEY: I think we'll want another subcommittee.

DR. PETTI: My biggest concern is not having the fuel section. I mean, will we add enough value without that volume? And it sounds like that volume is not two months away. It's further than that.

CHAIR BLEY: I don't disagree with you, but we'll have a shot at that later on.

MR. CASE: I don't know. I'm looking back to my fearless leader back there in advanced reactor land. Is that your thoughts as well, John?

MR. MONNINGER: Good afternoon. I'm John Monninger. I'm the Director of NRC's Division of

Advanced Reactors. Thank you very much for your time and discussions this afternoon. As was discussed earlier, last summer, the industry was in here and the staff presented in the fall, in the November time frame.

So when we think about things, I think as Mike indicated, we want to be prepared. There's tremendous interest in assuring that we are prepared. And we believe fundamental in that preparation is assuring that we are either aligned with ACRS or we understand where we are different from the ACRS and the reason for that.

So when we presented at the November meeting, whether we hinted at it or we explicitly requested, we were looking for a letter, whether that is this month, three months from now, whatever. But we definitely are investing considerable staff time and resources in this.

You talked about the NRC or the staff talked about the NRC potentially adopting codes being developed by the Department of Energy. That's a fundamental shift for us. It brings about a significant increase in our infrastructure also because we will continue to have the existing codes for the operating fleet plus there's additional resource implications of these new codes.

So we think it's important for whatever feedback there is, if possible, to have that feedback at the time that the ACRS would be ready. And we would be prepared to support additional meetings as needed.

The Congress has given the NRC significant funds for all the fee base. We expect to be reporting back to the Commission, to the Congress. I anticipate -- not that I would necessarily volunteer it. But I would anticipate that the NRC's Inspector General as part of the audit would be looking at, where is the NRC in terms of preparedness for advanced reactors? And the independent views from the ACRS are very important too. They've come for EP, security, LMP, et cetera. This is a big, big chunk of our resources and efforts. So that's sort of our plea.

CHAIR BLEY: We're going to have a discussion from the committee after this is all done.

MEMBER KIRCHNER: May I just ask --

CHAIR BLEY: Absolutely.

MEMBER KIRCHNER: -- including John? I'm looking at the last sub-bullet there. So where do you see yourselves? I mean, in the licensing modernization plan, this was an activity that was identified along with functional containment, emergency planning, et cetera. So how ready do you think you're going to be to actually apply these methods? And I think the first trial by fire will come with the source term -- mechanistic source term which is so important in establishing things like emergency planning, et cetera.

MR. MONNINGER: So the NRC has always said that we can review and potentially license non-light water reactor and advanced reactors now. It's a matter of how effective and efficient, how much time that's going to cost, and the amount of conservatism we've put within that analysis and the bounds on applicants. So I think when you said the last bullet up here with the LMP.

MEMBER KIRCHNER: Yeah.

MR. MONNINGER: So that was intended to reflect, I think, some of those discussions you had earlier this morning with regards to Steve's presentation and then the intersection with the afternoon. Where is the NRC going to decide to exercise DBA-type codes versus the more consequence-based LMP? Will we potentially gravitate to the frequence consequence curve in total and not do any type of design basis type calculations?

Or on the other hand, if the sequences -- the accident sequences within the LMP, will applicants come in and try to use a design basis curve as opposed to propagating out the potential consequences? So we haven't fully come to a conclusion where the intersection is between the DBA codes, where we will be using them in the future versus the consequence codes from the various accident sequences. So that's what the last one was intended to reflect. Will we go all the way to consequence only type independent calculations within the NRC?

MR. CASE: So that affects the path.

MR. MONNINGER: Yeah.

MR. CASE: So I bet you you're not smart enough today to make that call. So you sort of say, hey, I see a path I'm on --

MEMBER CORRADINI: That's never stopped us before.

(Laughter.)

MR. CASE: That path seems reasonable. Or there's enough information to say, hey, you really should be on an entirely different path.

MEMBER CORRADINI: But I want a clarification. What do you mean by a consequence only path?

MR. CASE: I think I know what you mean, but I'm not completely sure.

MR. MONNINGER: So if you were to look at the LMP and the frequency consequence curves for the anticipated operational occurrences for the design basis events or the beyond design basis events. Will we look at -- if you were to look at your typical fuel performance criteria, your staff tools, et cetera, will we even look at those types of surrogate metrics? Or will we just look at whatever that particular sequence is if it propagates some level of source term, even if it's coolant source term? Just look at that, and if it's acceptable, we're done.

CHAIR BLEY: We'll paraphrase you just a little.

MR. MONNINGER: Yes.

CHAIR BLEY: Because what it means is that a whole set of design basis accidents being treated in the traditional way --

MR. CASE: Would go away.

CHAIR BLEY: -- that would go away. That's where you're talking. Go ahead and finish.

MR. MONNINGER: Yeah, that's --

CHAIR BLEY: That's a possibility.

MR. CASE: -- the possibility.

MEMBER CORRADINI: But just if -- we're in discussion mode. But if that's what you're saying, then we're getting rid of a level of conservatism and I'll call it a margin because of uncertainty that we always put there because we're not really sure. And because we're not really sure, we think we know the population of all these groupings. It's not just one accident sequence but a grouping accident sequences and where they sit in a frequency dose calculation or consequence calculation.

And then there's this thing to the boundary. And we want to make sure we're far enough away from that boundary. And one of the ways to test it is to look for things that have to be safety related, things that have to be -- I'll use single failure criterion. But that's gone away in terms of the LMP. But something like that in terms of that I'm a good bit away from that boundary before I feel comfortable that life is okay. It seems to me that you still need a calculational tool set to do that.

MR. MONNINGER: And I think that's -- a counterargument would be within the LMP we are explicitly bringing within the licensing basis beyond design basis events where historically just a few of them had been added in. So in the past, if you had this level of conservatism, the deterministic approach due to the other -- and there's always going to be the unknown events. But the events that you didn't consider within your design basis and now we are bringing all of those in.

(Simultaneous speaking.)

MS. CUBBAGE: Sure. So I was just going to say it a little differently that it could be for some of these designs due to the inherent characteristics and some of the advanced safety features that we could see for design basis and even beyond basis significant margin to the dose limits which is ultimately what we care about, protecting public health and safety, such that we wouldn't feel it necessary to confirm those calculations.

The applicants would be responsible for doing all the analysis. But we would choose to look at scenarios where we're challenging the dose limits more so --

CHAIR BLEY: On a design-specific basis.

MS. CUBBAGE: -- on a design-specific basis. But we don't know yet where the designs are going to fall out, what the margins are going to be. So we're trying to do an across the board readiness activity. But really when push comes to shove and we have an applicant that's showing their dose consequences pegged at the axis, I don't know that we would spend the resources or time to affirm that with our own calculations. That's, I think, what John was trying to get to.

MR. MONNINGER: They would still have to do that.

MEMBER CORRADINI: I read it -- I don't mean to get ahead of you, but I read it differently. I read it but now I want to say it the way I thought I heard it which is I could have a DBA set of tools. I could have beyond DBA set of tools, so I don't need that. I just have a set of tools.

MS. CUBBAGE: Well, that's --

MEMBER CORRADINI: And then I take those set of tools and I do audit calculations what I think are necessary to make sure that the applicant is really where they think they are because we had some sort of evaluation procedure to do that. That's what I thought I heard you say.

MR. MONNINGER: And it was some of the terminology issue of the historical baggage with DBAs.

MS. CUBBAGE: The extent to which we can come up with one tool that can cover this. So could we use MELCOR for all scenarios? I don't know that we thought that.

CHAIR BLEY: So when we get to our committee discussion after this is over, it sounds to me like there are -- if we want another meeting, there are a variety of things we might look at. Some are kind of almost policy-like things and other ones are more detail in some of the things we heard today. And we have to, over the next few weeks, decide what we want there. Anything more?

MEMBER CORRADINI: No, that's helpful.

CHAIR BLEY: Mike?

MR. CASE: That's it. That was fantastic. Thank you.

CHAIR BLEY: Thanks to you and to the whole staff. Do we do public comments first? My brain's gone dead. We're going to do public comments, both on the phone line and here, if you can make sure the phone line gets open. Is there anyone in the room -- oh, wait. I'm sorry. We were supposed to have a statement from Dave Henderson. Is he here? As a public comment. Yeah, I'm sorry. I forgot already.

MR. HENDERSON: So thank you. So I'm Dave Henderson. I'm with Department of Energy and the Office of Nuclear Energy. I'm the program manager for the mod-sim programs, both the hub that does light water reactor and NEAMS that has the advanced reactors. So for one, I wanted to make sure you were aware. You should've received a letter from the Deputy Assistant Secretary Shane Johnson.

CHAIR BLEY: We did, and that'll be attached to the minutes of this meeting.

MR. HENDERSON: So just for the public record, I thought I'd summarize a little bit of that, some major statements from the DOE side since we've been talking about some of these codes, especially in the first part.

The first being that DOE is fully supportive of NRC's plan to use the DOE codes, whatever codes are most useful to the NRC. We're fully supportive of their CRAB vision. From our perspective, we see it as an addition return on the significant investment that DOE has made and in these codes over the years.

But even more so now, it's a success metric to us. So we are interesting in supporting NRC however we can. Whether that be training that we've already started to provide or are committed to providing in the future. V&V which this Volume 1 report I think does a great job of outline what the NRC sees as the needs. And we're starting to use that as a discussion point for what makes most sense for DOE to do, NRC, and/or industry and so how we move forward there.

Quality assurance for sure. Access to the codes and to computing resources as needed. So this will be a priority for us. If NRC says they need something within the lab system and we have it, we'll work to make sure that it happens. We have been working with NRC, Office of Research especially, but more broadly as well on who should be doing what, what's needed overall and not just in the V&V space.

And so I just want to make those brief statements and refer you to the full letter for some other specifics as you need.

CHAIR BLEY: Can I ask you a question?

MR. HENDERSON: If you like.

CHAIR BLEY: Yeah. I didn't know if you have access to Volume 1.

MR. HENDERSON: Yes.

CHAIR BLEY: And you have. In the areas where they've described -- the staff has described gaps and levels of maturity of the codes for the different reactor types, are you thoroughly in agreement with that or have you had time to really sort that out?

MR. HENDERSON: I'd say it's a little bit early to make definitive statements. But we have looked through it. The analysis or the assessment of the maturity is a fairly qualitative and granular assessment. We don't feel like where they've identified gaps I think in most places, we're not of a very strong different opinion. So we feel like it's well founded. Whether we put it at one or two or vice versa or things like that, we really haven't gone into, and that'll be part of the ongoing discussion.

CHAIR BLEY: Okay. That is qualitative, but it's fairly descriptive and --

MR. HENDERSON: Yes.

CHAIR BLEY: -- qualitative in nature.

MR. HENDERSON: No, we thought it was a very good report and appreciate it.

CHAIR BLEY: Thank you. Are there any other comments from people in the room? Come to the table. Anybody else? Name and organization and your comment.

MR. HAUGH: Thank you. My name is Brandon Haugh. I'm the Director of Modeling and Simulation at Kairos Power. We're in pre-application discussions with the NRC for our review that's going to begin very soon. We're going to be using some of the tools out of the NEAMS toolbox that are on that graph and we just wanted to reinforce that it's coming very soon.

And we realize it's a stage process where we're going to be coming in to talk about what we intend to do with those codes to get feedback before we go off and spend significant resources on detailed testing. We want to get some alignment there on what the validation scope should be so that we spend money in the right places.

I also wanted to discuss a little bit about Dave Petti's discussion. The sky isn't falling. We operate graphite reactors all over the world, also very high enriched reactors well about five percent all the time. So some of that is a little bit -- when you get down in the weeds and look at one specific thing and perturb it, it's just like a lattice code, right? I can go to a lattice code and go all the way down into the pin and I can look at solution variations around how do I treat the mesh and all that. In the end, when you look at integral reactor problems, many of those things are not that important.

So I just don't want to get down to areas where we feel we need to spend decades of doing research on something that's already operated. We need to move the advanced technologies for the United

States to support a healthy nuclear industry and to compete globally.

CHAIR BLEY: Thank you. Anyone else in the room? The phone line is open. Is there anyone on the phone line who would like to make a comment? If so, identify yourself, please, and make your comment. I guess not. It was a long time to stay on the telephone. Okay.

Well, I would like to thank everyone for today's presentations. At this time, I'm going to ask just to go around the table, get comments from the members and from our invited expert. And Professor Ballinger, would you go first, please. And also thinking about things we would want to hear about before we got to the point of writing a letter in the not too distant future.

MEMBER BALLINGER: I thought the material that we provided was very good. The presentations were actually a good augmentation of the material which we read which was a lot.

I'm kind of not a reactor physics person or anything. But when I hear the discussion going back and forth where sometimes we get down into the weeds about how much pcm difference we have and then somebody say, well, it doesn't really make a whole lot of difference. I'm wondering whether we shouldn't make sure that we answer the question, what is good enough, in each of these cases. What is good enough?

And then find the gaps that prevent us from doing what's good enough. And if it means using a DOE code because we absolutely need it, fine. If it means using a code that we already have here that we can modify, that's fine too. But I wouldn't think that we would just arbitrarily go to some of these CASL codes and stuff like that. You can elephants fly. They probably do. But that would be my comment, to try to get through this by making it good enough.

CHAIR BLEY: Thanks, Ron. Dave?

DR. PETTI: Yeah, I also thought the presentations were really helpful. The one thing that I mentioned earlier in my questions was in my mind separating what I call physics and chemistry uncertainty from code augmentation and functional capabilities. They're very different in terms of the timelines it takes to implement.

And I think if you apply that metric, you'll find for each family of reactor technologies very different answers in terms of where you are and what you have to do. In my mind, it's a lot different to say, yeah, I need to make this 3D over here and my code is only 2D or I don't have the data that I need. They take you in very different distances.

I think you have to do a lot of sensitivity analysis once you get something that you feel is together and good enough to help inform your thinking. My favorite example was when I was here on the other side of the table of NGNP and pounded by temperature uncertainties in pebble beds and in prismatic reactors and all the thermohydraulics. And I left it going, well, there's an easy way to fix that. I'll just test the fuel 300 degrees above what the designer thinks is the worst thing. That's what we did.

So we took that. That's off the table. I mean, it'll come back. But the answer is the fuel is fine up there. So there's more than one way to skin the cat. Sometimes the sharper pencil you need. Sometimes you don't, right? But you just have to identify really where the issues are.

What I do worry about is getting overly simplistic. You've got to find a balance here. If you put the designer hat on and you guys come up and say, no, I don't agree with your calculation. This is what I get with my code. And the answer is your code is too simplistic. You're not capturing it.

We're not pushing the licensing -- we're not expediting the licensing. We're going backwards, right? Because now you'll get in a discussion and you'll have to figure out where the difference is. And if the difference was you oversimplified when the better tool could've helped you, we've got to keep that -- we've got to figure out that balance, I think.

CHAIR BLEY: Thank you. Dick?

MEMBER SKILLMAN: I want to thank the staff and all the presenters for a very thorough afternoon. I have no more to add. Thank you.

CHAIR BLEY: Matt?

MEMBER SUNSERI: Thanks for the presentation. I thought they were informative and useful. I think when I think about the licensing modernization and streamlining and using other people's codes, the Department of Energy or even the licensee applicant, I always harken back to I think one of the strengths of the industry is probably not often stated this way. But I think one of the strengths of the industry has been the fact that's it's overseen by a strong regulator.

When things go bad, the question that's usually asked, where is the regulator on this thing? We're seeing it in the airline industry right now.

I've seen it in the chemical -- we've had a couple of chemical fires in Texas and the questions about where the regulator stands.

So I think when it comes down to the validation, even if the margin is big, I think the regulator needs to be able to say, yes, and we validated that in a very demonstrative way. Whatever that is, I think all these approaches probably leads to that. But I think having that ability to do that independently is going to be important for the future and the industry is going to want that. So that's all I had.

CHAIR BLEY: Thank you. I think while we're here, Jose? Just catching you off guard.

(Laughter.)

MEMBER MARCH-LEUBA: Yes. I wanted to say that one of the primary value of the NRC review of the -- the guys have designed the reactor. There was another reactor. But the primary value of the NRC review is to find out why they miss. Is there something important? And a particular physics phenomena or a particular transient that they didn't think of. Anytime we have a severe accident is because something happened that we didn't anticipate. So by having a second pair of eyes looking over their shoulder, we are providing a lot of value.

Now that said, you don't need to do the super detailed calculation with 100,000 cores calculation for a week. And that's why they keep pushing for the steady state simulator for DBA. Because with a steady state simulator and some slice calculations, you can bind most of your transients.

And I say it jokingly before. But except for the microreactors, if they don't have a factor of two to ten to SAFDLs, they shouldn't bother to submit it. These reactors are going to have a tremendous amount of margin. So checking out that you don't exceed it because of something unusual that they didn't think of is what we are supposed to do.

And in that sense, the other concern I have, if we go with the full CRAB simulation that we -- the platform that we're proposing, I'm concerned all the staffing and the amount of time it will take to do inventories. Really we're all cool guys. We're all physicists. You always underestimate how long you're going to take. And there's going to be only two or three guys who know how to run and they're going to move to Florida before the reactor comes in.

So that's it. Next.

CHAIR BLEY: Walt?

MEMBER KIRCHNER: I'll just add my thanks to the staff. Thank you very much. I am pondering this sub-bullet up here, and that's where my thinking is. I'm mentally putting myself through doing a mechanistic source term calculation for one of these advanced reactors as a readiness test. So if we have another meeting, I would like to explore that topic in more detail.

CHAIR BLEY: Send me a --

MEMBER KIRCHNER: Thank you.

CHAIR BLEY: -- paragraph on that.

MEMBER KIRCHNER: Okay.

MEMBER CORRADINI: So thanks to the staff. We've been thanking them the whole day. But I think they did a nice job of leading us through this. I'm still back to my attributes, principles. And they come down to -- I think Dave said, physics and chemistry. I call them physics gaps. What are the things that we don't know? What do we actually have to work on to do a better job?

I still think we need a minimum set of tools, and I think we need phase development and validation. And what I mean by phase development is we can't do all five types of reactors or four, whatever. I lost track of how many classes of these things are.

We've got to make some sort of decision as to what's going to hit you first. And it doesn't have to be necessarily NRC licensing. And the current world we live in, the NRC may not license it. But they may be asked to provide technical support to somebody else who says they're already got it running in terms of Defense Department or whatever.

So I think NRC has got to be ready for that. So you've got to decide which of these you want to start with to develop a pilot process on how you're going to use your minimum set of tools. And then back to validation because some of these things are close enough to validation, you may not need to this, where others you're years away. So those, to me, are the principles.

The one thing that took me by surprise is what Jon and Amy had said. So I don't know what to do about that other than we're going to have to talk about it with the committee. I think Walt's hit upon it. I took these four questions and it comes down to working the problem backwards. If source term is all we care about for safety, then maybe that's the problem we work and we work backwards.

MEMBER KIRCHNER: Well, this gets into a

discussion, but I can't let it go --

CHAIR BLEY: It's all right.

MEMBER KIRCHNER: -- with just that. You still go back and do conventional deterministic analyses on key phenomena to just validate or confirm the key design --

CHAIR BLEY: You can't do a probabilistic analysis without that kind of --

MEMBER KIRCHNER: Without that item. So it's not an --

CHAIR BLEY: It's not a good one.

MEMBER KIRCHNER: It's not an either/or.

MEMBER CORRADINI: No, it's not. But it's working the problem from what you think you have to do with these advanced designs to what you'd like to do.

CHAIR BLEY: Joy?

MEMBER REMPE: I have a couple of points. The first one's a quick one just to reiterate the comments I made earlier about considering a broader spectrum of events. I think the PIRTs focused on certain operational concerns that AOOs, DBEs, the beyond design events. And I think you may end up with something where, again, transportation, installation of a module may be a more significant contributor like your tools can identify that risk.

The other thing, and that's why I kind of broke in when I heard what you were thinking about you want a letter now. I think the thing about integration, I learned a lot today from the presentations that I couldn't get from the documents.

And I think words might help with what Mike and Walt have said. And also be more consistent with the licensing modernization approach where you get rid of this boundary between beyond design basis events and design basis events. And if you think about that you're going to have an integrated tool that would help you with the LMP you would use, it is a bit more simpler.

And then rely on the more detailed codes in a way that we did with consequential steam generator tube rupture where you use CMD (phonetic) to look at the detailed phenomena. And it doesn't have to be an NRC approved code that's maintained in house. But it's just something you might draw upon as needed and then extract parameters that can be put in to the more simplified code.

And again, if the documents were more integrated and you got rid of that design basis and beyond design basis. I think some of my concerns that I had coming into this meeting would've been alleviated and also then talking to each other about how you're going to compare some of the results with the benchmark of the data that are available and make sure that those two documents are consistent would help me.

And then finally, I still remain very -- well, I also wanted to comment that I was really happy to hear the source coding is available. Because if you are going to have those tools, things come and go as we've seen in U.S. history with funding from the EOB. And so if you're going to invest anything, I'm glad to know that you have something that you'll have at the end of the day.

And I would hope -- but I still am concerned about the cost for getting those tools validated. And I think your interactions with DOE should be very beneficial to the department, and I hope that they're listening to the data needs that you've identified and that they take action as appropriate as they can with funding to do something about those data needs.

Because I think that will -- again, you'll have to do some prioritization and to identify which things are most important. But I think that that is a good use of our tax dollars in general for the U.S.

that you guys are doing that with this fee base funding.

But I think that's the major points I wanted to emphasize. And again, I appreciate everybody's work and presentations.

CHAIR BLEY: I too want to thank you for the presentations and discussion and kind of come back on something I think Walt said. There was a lot of things we heard today that's not in the two volumes. We're going to take some time to figure out what we want to hear more about before we write a letter for you.

I think it's a couple weeks till we get the transcript back. Weidong and Derek, as soon as the transcript is available, please give it to all the subcommittee members and also start thinking about looking for a data a couple months away maybe for a subcommittee and then right after that a full committee meeting. And I'll talk with you guys later.

For the members, this kind of sinks in a little bit. Send me any notes you have about things we ought to pick up and if we have another meeting or if we don't need another meeting, whatever you think, and agenda items for such a meeting.

I guess that's about it. We'll be in

touch through Weidong and Derek about when we want meetings and what we want to talk about. Or maybe we'll have another informal about agendas and the like. So thanks to everyone and we are adjourned.

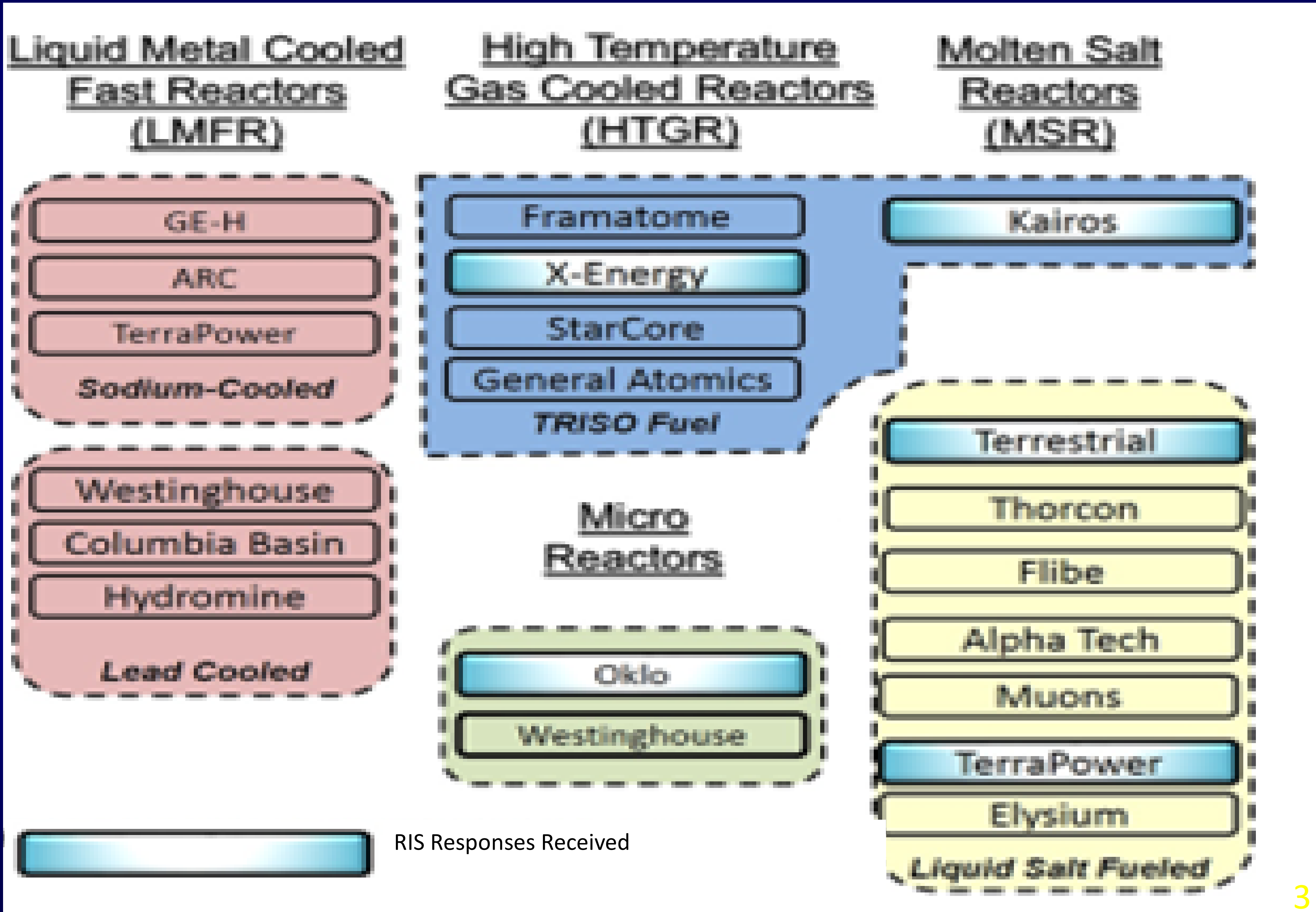
(Whereupon, the above-entitled matter went off the record at 6:07 p.m.)

Evaluation of Codes for NRC's Regulatory Oversight of Non-Light Water Reactors

Kimberly A. Webber
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission

Presented to:
Advisory Committee on Reactor Safeguards
May 1, 2019

Non-LWR Landscape



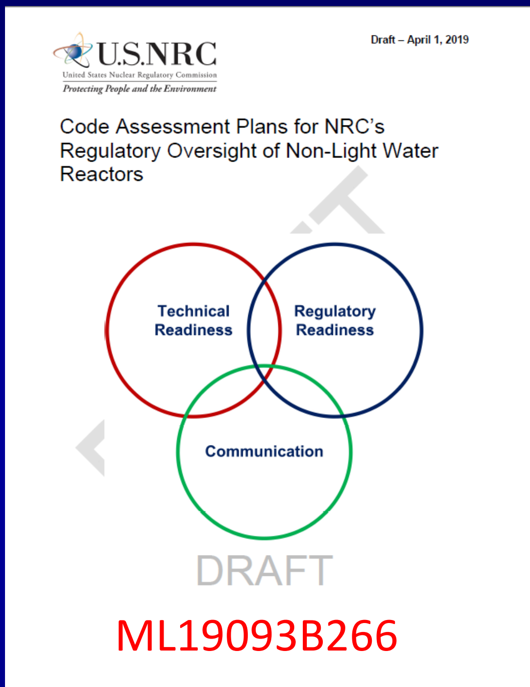
Using NRC's Resources Efficiently

- Pursuing flexible codes for use with a wide variety of reactor technologies
- Initiating tasks requiring long lead time data needs (e.g., molten salt reactors)
- Promoting staff's understanding of potential applicant's analytical tools and methods
- Identifying database needs and obtaining data applicable to multiple designs
- Minimize costs by leveraging DOE tools and coordinating development plans

Accomplishments

- Preliminary gap analyses to identify code development needs
- Evaluation of NRC, DOE, and commercial code capabilities for applicability to non-LWRs.
- Staff participation in DOE training for non-LWR technologies and in the use of NEAMS codes
- Preliminary code development & assessment activities
- Detailed code development plans

NRC's Implementation Action Plan, Strategy 2 – Computer Codes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United States Nuclear Regulatory Commission
Protecting People and the Environment

Draft – April 1, 2019

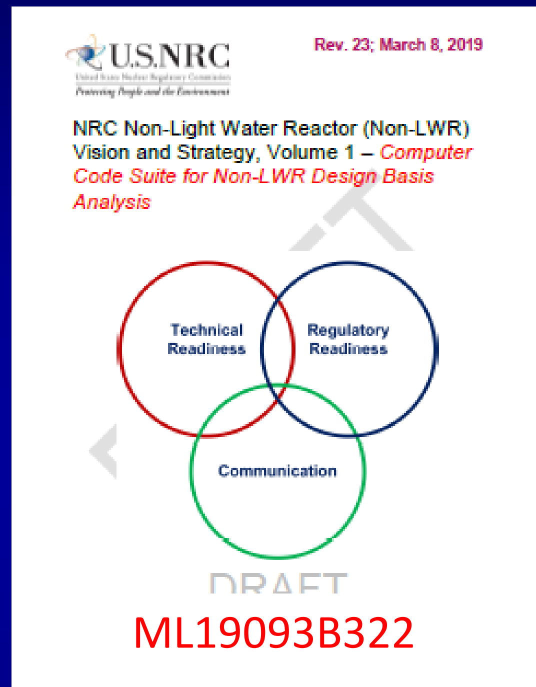
Code Assessment Plans for NRC's Regulatory Oversight of Non-Light Water Reactors

Technical Readiness
Regulatory Readiness
Communication

DRAFT

ML19093B266

Introduction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Protecting People and the Environment

Rev. 23; March 8, 2019

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 1 – *Computer Code Suite for Non-LWR Design Basis Analysis*

Technical Readiness
Regulatory Readiness
Communication

DRAFT

ML19093B322

Volume 1


U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy

Volume 3: *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

Technical Readiness
Regulatory Readiness
Communication

ML19093B404

Volume 3

Volume 2 = Fuel Performance
Volume 4 = Radiation Protection

Under Development

Modeling and Simulation of non-LWRs for "Design Basis Events"

Stephen M. Bajorek and Joseph Kelly
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission
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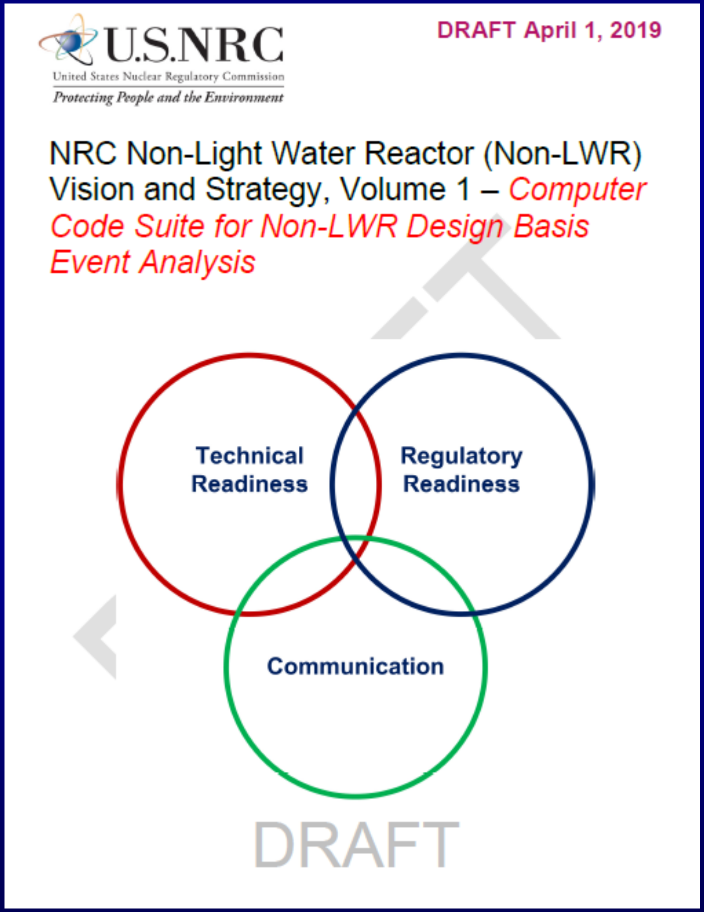
FUTURE PLANT DESIGNS SUBCOMMITTEE BRIEFING
May 1, 2019

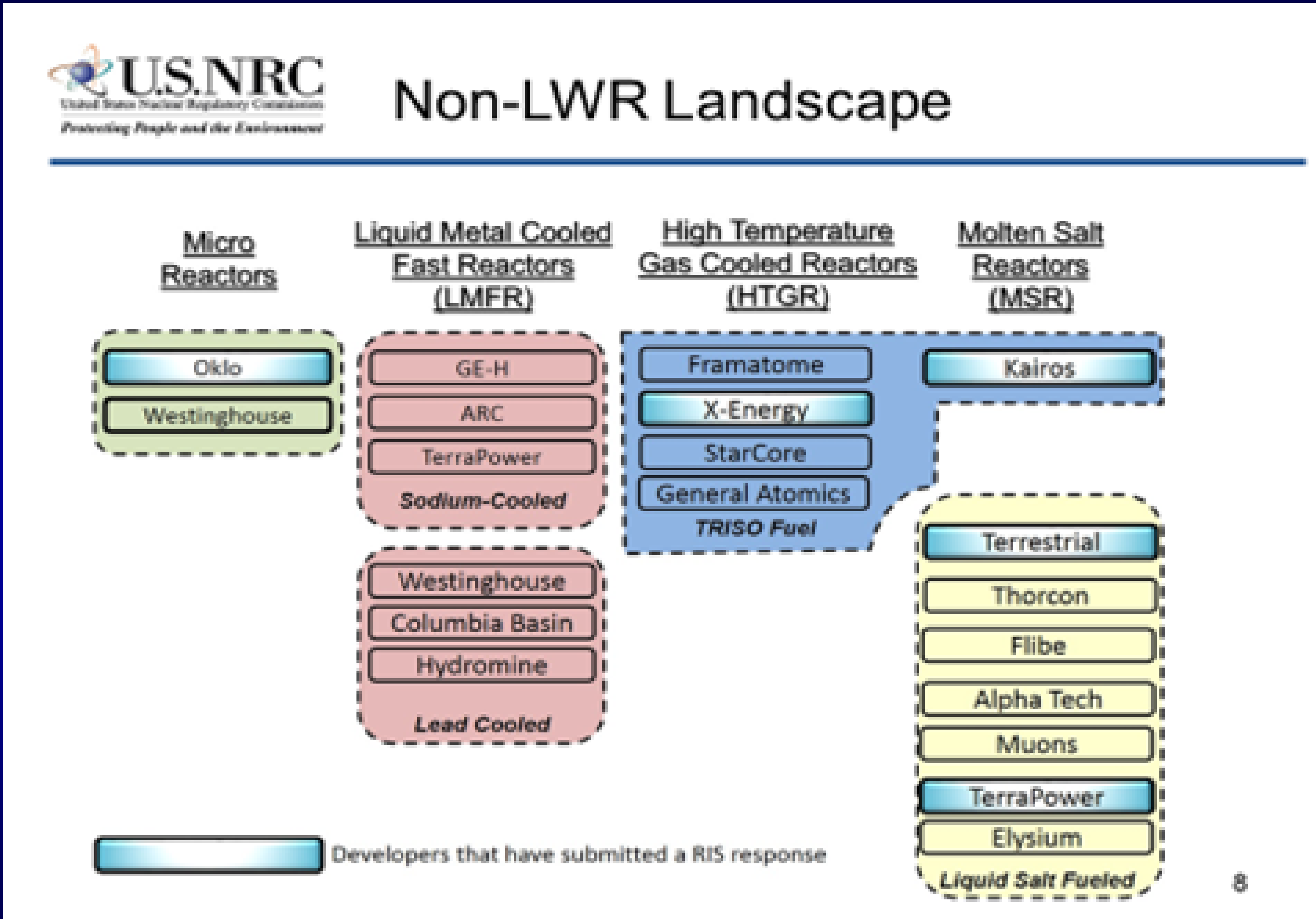
Event Selection

- “Chapter 15” vs “Chapter 19” deterministic approach to be replaced with PRA insights.
- “Design Basis” Code(s) = those to be used for confirmatory analysis of events that little/no core (geometric) disruption or fission product release.
  - Unprotected loss of flow
  - Loss of heat sink(s)
  - Events that may involve multiple failures
- “Beyond Design Basis” Code(s) = for events involving core melt, fission product release & transport.

Introduction / Outline

- Volume 1 “Design Basis Event Analysis” :
  - Phenomena Identification and Ranking Tables (PIRTs)
  - Event Scenarios
  - “New” Physical Phenomena for non-LWRs
  - Gaps
  - Tasks

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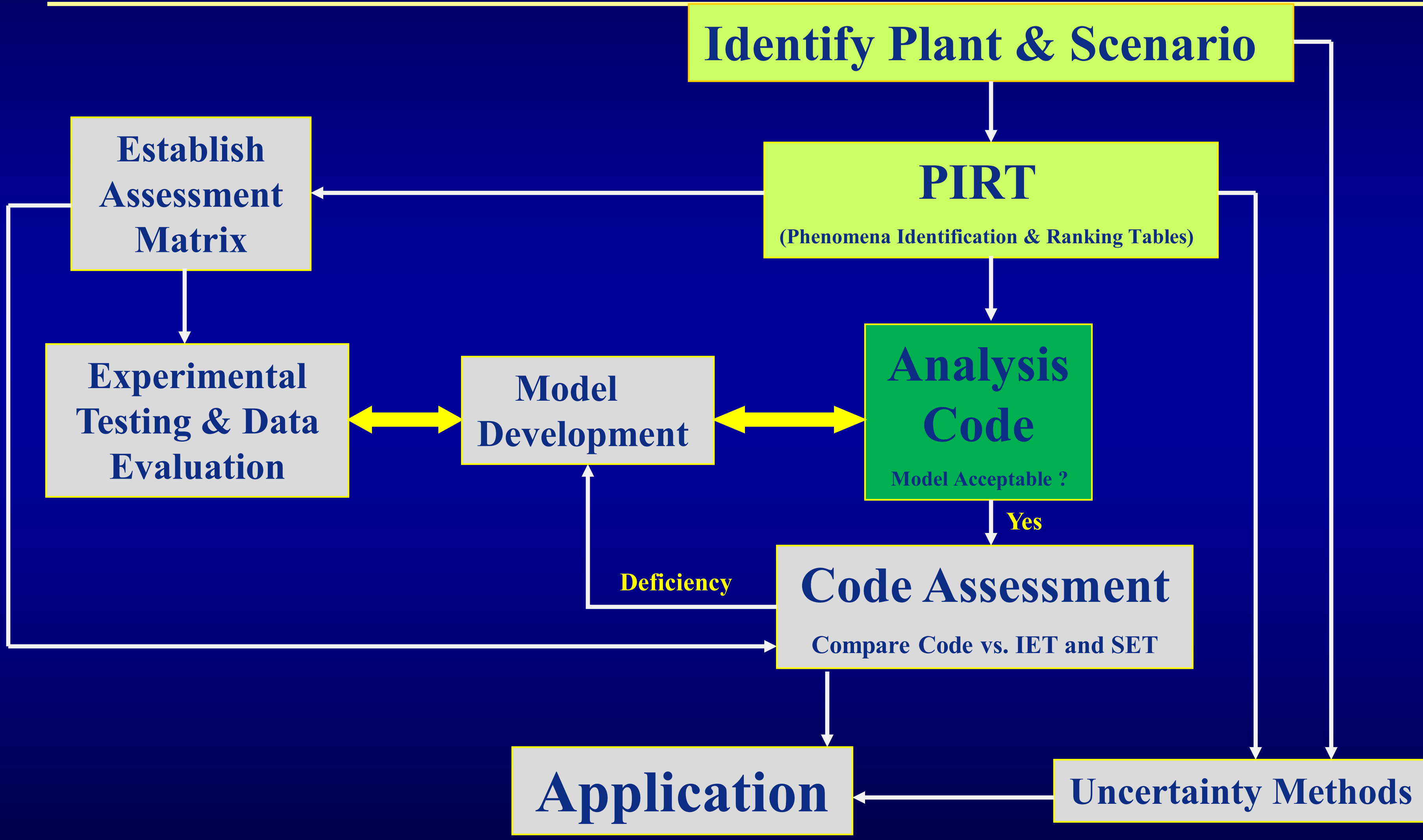
Non-LWR Landscape
Micro Reactors
Liquid Metal Cooled Fast Reactors (LMFR)
High Temperature Gas Cooled Reactors (HTGR)
Molten Salt Reactors (MSR)
Oklo
Westinghouse
GE-H
ARC
TerraPower
Sodium-Cooled
Westinghouse
Columbia Basin
Hydromine
Lead Cooled
Framatome
X-Energy
StarCore
General Atomics
TRISO Fuel
Kairos
Terrestrial
Thorcon
Flibe
Alpha Tech
Muons
TerraPower
Elysium
Liquid Salt Fueled
Developers that have submitted a RIS response
8

Characterization of Design Types

Plant Type No.	Description	Fuel
1	HTGR; prismatic core, thermal spectrum	TRISO (rods or plates)
2	PBMR; pebble bed core, thermal spectrum	TRISO (pebbles)
3	GCFR; prismatic core, fast spectrum	SIC clad UC (plates)
4	SFR; sodium cooled, fast spectrum	Metallic (U-10Zr)
5	LMR; lead cooled, fast spectrum	Not available. (Possibly nitride fuel.)
6	HPR; heat pipe cooled, fast spectrum	Metallic (U-10Zr)
7	MSR; prismatic core, thermal spectrum	TRISO (plates)
8	MSPR; pebble bed, thermal spectrum	TRISO (pebbles)
9	MFSR; fluoride fuel salt, thermal/epithermal spectrum	Fuel salt
10	MCSR; chloride fuel salt, fast spectrum	Fuel salt

Reg Guide 1.203 Code Development



Phenomena Identification and Ranking Table

- Reg Guide 1.203 process begins with PIRT and scenario definition.
- Several PIRTs are available, and have been used to help identify important phenomena and "gaps" in the technologies.
- Initial focus of staff efforts were on molten (fuel) salt reactors. A "pre-PIRT" was developed to identify important processes. These will be re-considered as designs are finalized and submitted.

PIRT References

- [6] S.J. Ball and S.E. Fisher, Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), ORNL/TM-2007/147, NUREG/CR-6944, Vol. 1, March 2008.
- [7] USNRC, TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents," NUREG/CR-6844, 2004.
- [8] R. Schmidt, et al., "Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety," SAND2011-4145, 2011.
- [9] Lap-Yan Cheng, Michael Todosow, and David Diamond, "Phenomena Important in Liquid Metal Reactor Simulations," BNL-207816-2018-INRE, ADAMS Accession No. ML18291B305 (2018).
- [10] Hsun-Chia Lin, et a.l. , Phenomena Identification and Ranking Table Study for Thermal Hydraulics for Advanced High Temperature Reactor, *Annals of Nuclear Energy*, 124 (2019) 257–269.
- [11] Farzad Rahnema, et al.,, "Phenomena identification and categorization by the required level of multiphysics coupling in FHR modeling and simulation," *Annals of Nuclear Energy*, 121 (2018) 540–551.
- [12] D. J. Diamond, N. R. Brown, R. Denning and S. Bajorek, "Phenomena Important in Modeling and Simulation of Molten Salt Reactors," BNL-114869-2018-IR, Brookhaven National Laboratory, ADAMS ML18124A330, April 23, 2018.
- [13] Diamond, D. J., Brown, N. R., Denning, R., Bajorek, S., 2018. "Neutronics Phenomena Important in Modeling and Simulation of Liquid-Fuel Molten Salt Reactors," Advances in Thermal Hydraulics (ATH 2018); Nov. 11-15; Orlando, Florida, USA: American Nuclear Society.
- [14] Bajorek, S., Diamond, D. J., Brown, N. R., Denning, R., 2018. "Thermal-Hydraulics Phenomena Important in Modeling and Simulation of Liquid-Fuel Molten Salt Reactors," Advances in Thermal Hydraulics (ATH 2018); Nov. 11-15; Orlando, Florida, USA: American Nuclear Society.
- [15] J. W. Sterbentz, J. E. Werner, M. G. McKellar, A. J. Hummel, J. C. Kennedy, R. N. Wright, J. M. Biersdorf, "Special Purpose Nuclear Reactor (5 MW) for Reliable Power at Remote Sites Assessment Report; Using Phenomena Identification and Ranking Tables (PIRTs)," INL/EXT-16-40741, Revision 1, April 2017.

PIRT and Scenarios: GCRs

Scenarios

- pressurized loss of force cooling (P-LOFC) accident
- de-pressurized loss of force cooling (D-LOFC) accident
- reactivity-induced transients, including ATWS events
- (Events that involve air-ingress and significant oxidation of the graphite, water-ingress, transport and release of “graphite dust” will be simulated with a traditional severe accident code.)

Phenomena

- core coolant bypass flows,
- power/flux profiles,
- outlet plenum flows,
- reactivity-temperature feedback coefficients,
- emissivity for the vessel and reactor cavity cooling system,
- reactor vessel cavity air circulation and heat transfer, and
- convection/radiation heating of upper vessel.

PIRT and Scenarios: SFRs & LMRs

Scenarios

- loss of coolant with and without scram
- loss of forced flow
- unprotected loss of flow
- unprotected loss of heat sink
- reactivity-induced transients, including ATWS events

Phenomena

- single phase transient sodium flow
- thermal inertia
- pump-coast down
- sodium stratification
- transition to natural convection core cooling core flow
- decay heat generation
- reactivity due to mechanical changes in core structure
- reactivity feedback at high power

PIRT and Scenarios: "Micro Reactors"

Scenarios

- loss of heat sink
- inadvertent reactivity insertion transients, including ATWS events
- localized heat pipe failure
- cascading loss of heat pipes
- seismic event (causing reactivity increase)
- events related to coupling the reactor to the power conversion unit
- monolith temperature and stress under normal operating conditions
- monolith temperature and stress under postulated accident conditions.

Phenomena

- monolith thermal stress
- single heat pipe failure
- machining and inspection of the monolith
- heat pipe performance
- reactivity and core criticality

PIRT and Scenarios: MSRs

Scenarios

- loss of forced flow
- unprotected loss of flow
- inadvertent reactivity insertion transients, including ATWS events
- loss of coolant
- over-cooling events (leading to partial solidification)
- station blackout
- loss of heat sink

Coolant Salt MSRs

Phenomena of Concern

- thermophysical properties of coolant salt (conductivity and viscosity)
- wall friction in the core
- core flow asymmetry
- upper and lower plenum mixing
- safety system component performance
- chimney natural circulation and performance

Fuel Salt MSRs

Phenomena of Concern

- delayed neutron precursor motion
- salt chemical composition
- neutron absorption in fuel salt
- physical properties
- convective heat transfer
- primary system flow resistances
- structural material performance (swelling and expansion)
- tritium production and transport (fluoride salts)

"Modeling Gaps" Identified by PIRTs

- Phenomena that are significant and "new" with increased importance for non-LWRs relative to conventional LWRs include but are not limited to:
  - Thermal stratification and thermal striping
  - Thermo-mechanical expansion and effect on reactivity
  - Large neutron mean-free path length in fast reactors
  - Transport of neutron pre-cursors (in fuel salt MSRs)
  - Solidification and plate-out (MSRs)
  - 3D conduction / radiation (passive decay heat removal)

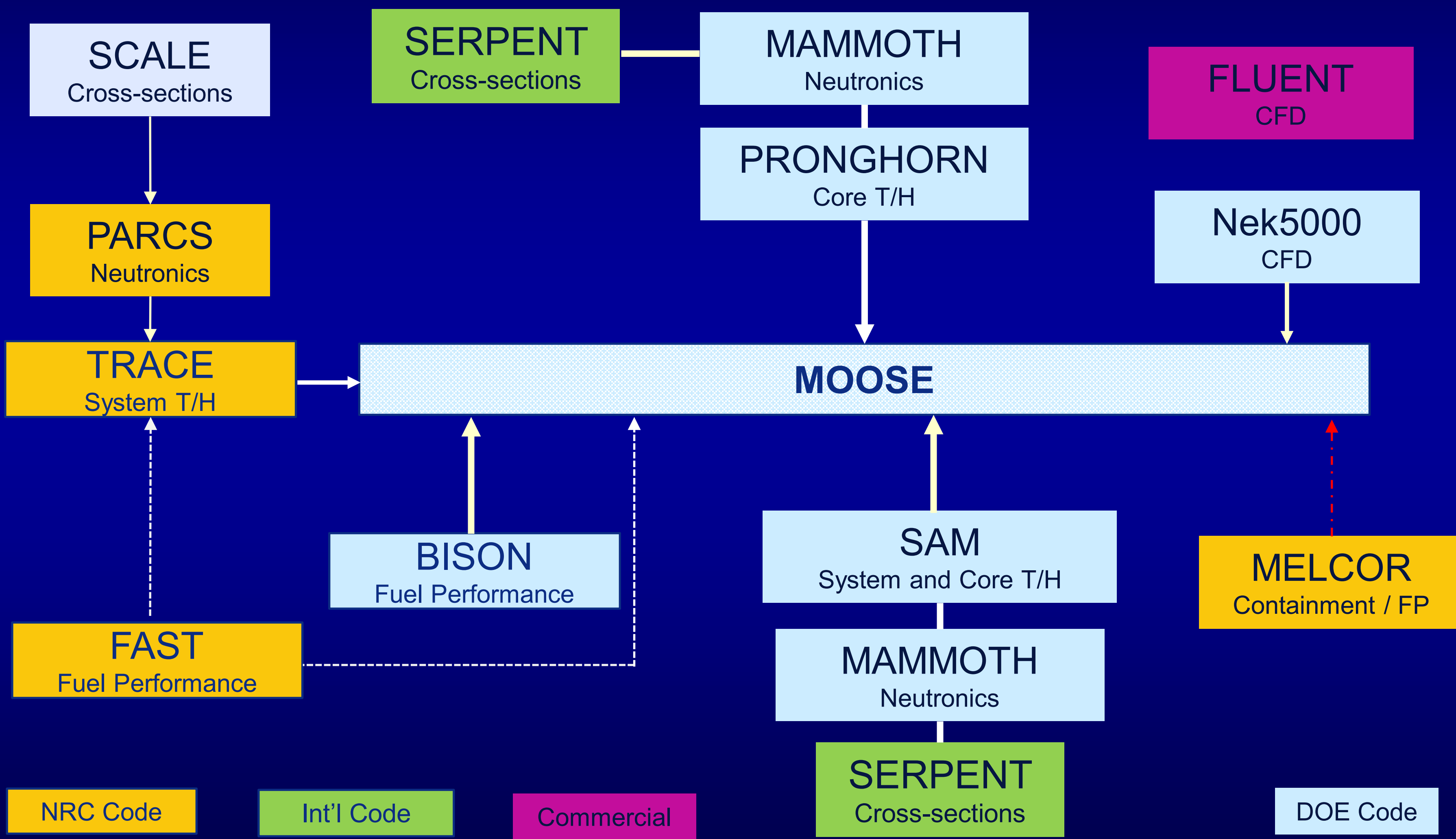
"Modeling Gaps in NRC Codes"

Codes for Design Basis Event Analysis

- Codes considered:
  - NRC codes (TRACE, PARCS, FRAPCON, FAST)
  - DOE NEAMS codes (MAMMOTH, PRONGHORN, RELAP7)
  - ANL codes (SAS4A/SASSY, SAM, PROTEUS, Nek5000)
  - DOE CASL codes (MPACT, CTF, BISON, MAMBA)
  - Commercial codes (FLUENT, COMSOL)

- Recommended approach is to use a system of coupled codes, “Comprehensive Reactor Analysis Bundle” (CRAB).   This includes codes from both NRC and DOE.

Comprehensive Reactor Analysis Bundle (CRAB)



Code Selection Considerations

- Physics. Code suite must now or with development capture the correct physics to simulate non-LWRs. Selection of codes based on results of PIRTs. Code coupling necessary for "multi-physics".
- Flexibility. Multiple reactor design concepts require flexibility within code suite. A goal has been to limit the number of new codes and need for staff training.
- Code V&V. Code validation is critical and represents the major gap in EM development. Database is weak for some designs.
- Computation Requirements. Must be able to run simulations on NRC desktops or HPC platforms readily available to NRC.

Codes selected for CRAB satisfy these criteria.

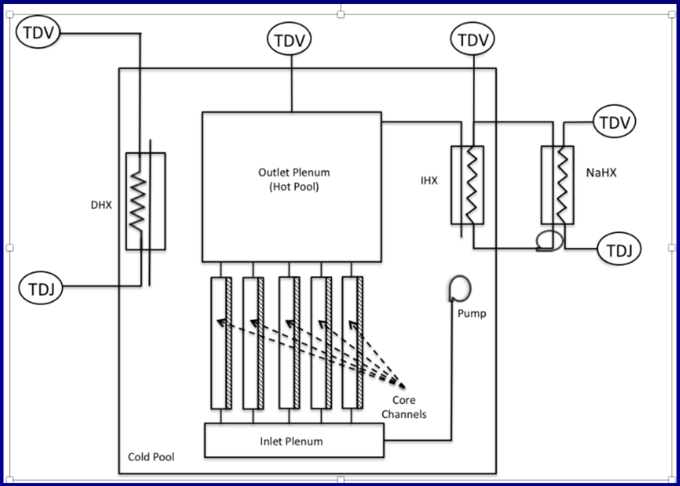
Unique Capabilities Available in CRAB

- Examples
  - Multiphysics Coupling
  - Geometric Fidelity
  - Advanced Equivalence Methods
  - Multi-Scheme Capability
  - 3D Reduced Order Flow Model

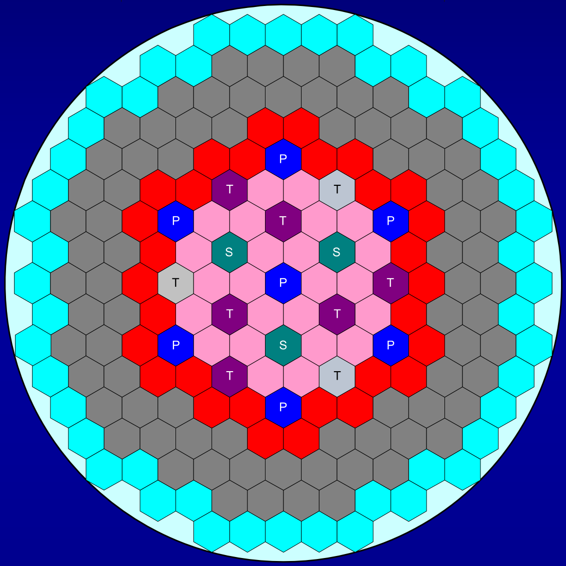
Goal: Enable analysis of advanced designs without over-simplifying assumptions to provide intermediate fidelity model for modest computational resources.

Multiphysics Coupling

SAM: System Level Thermo-Fluids



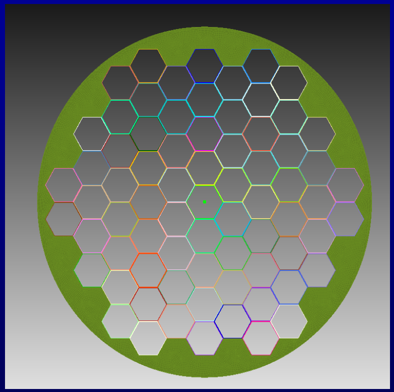
MAMMOTH: Rx Kinetics



Temperatures & Densities

Power

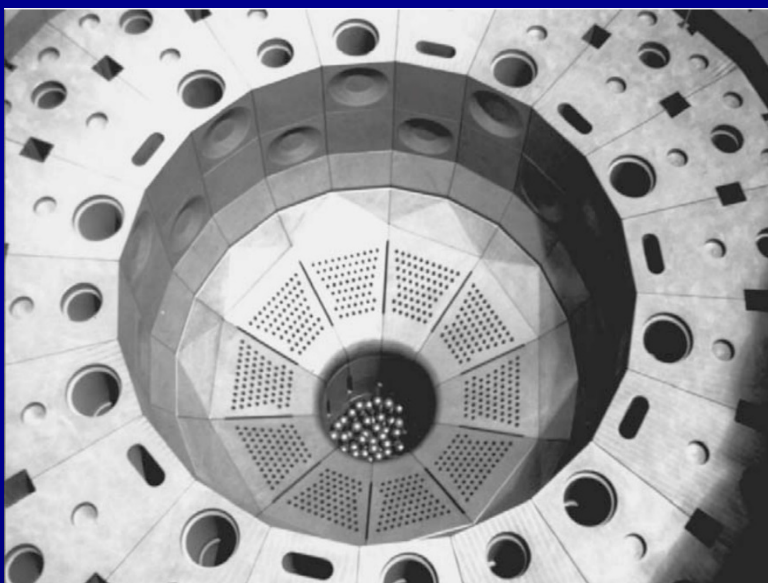
Temperatures



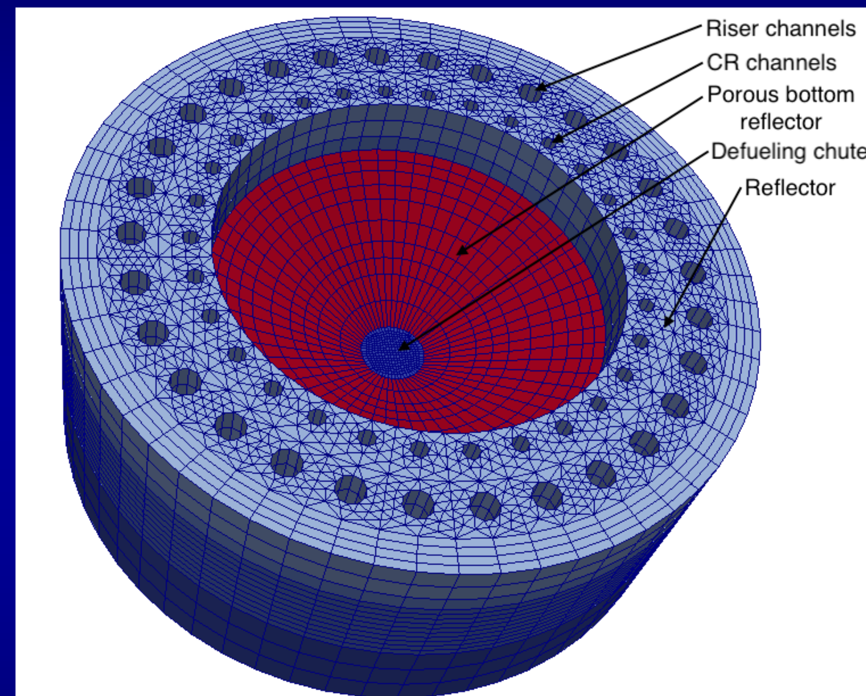
Displacements

Tensor Mechanics Module

Geometric Fidelity



HTR-10
Bottom Reflector & Conus



PRONGHORN Mesh
- 1-D Channels for upcomer & CR's
- 3-D Porous body for conus

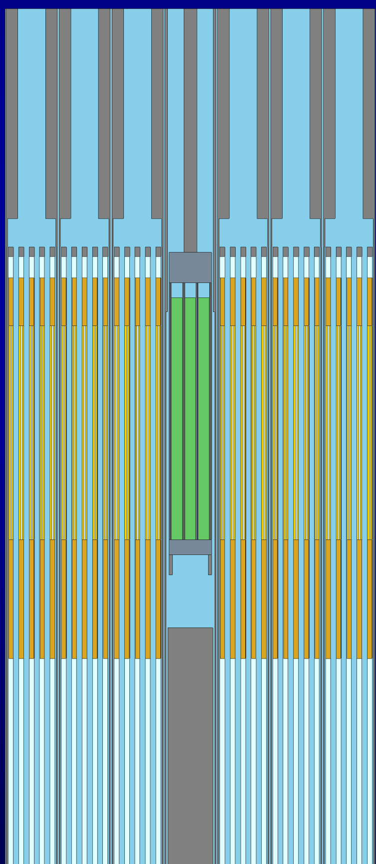
Advanced Equivalence Methods

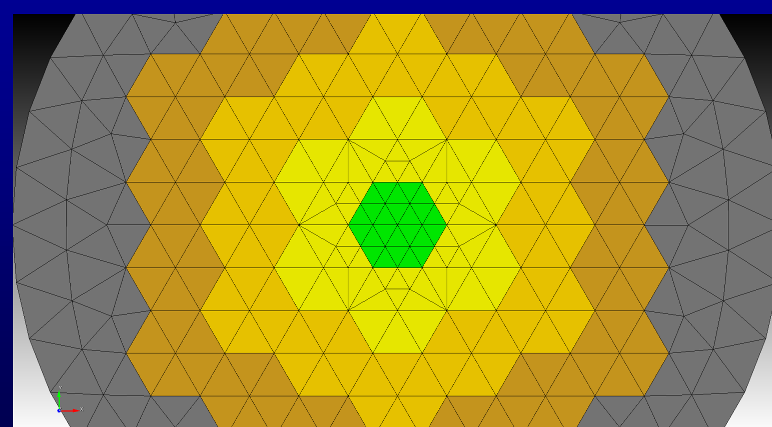
Monte-Carlo
- Compute multigroup cross-sections
- Compute reference quantities of interest (eigenvalue, reaction rates, leakage rates…)
- Currently done in Serpent

Equivalence
- In Rattlesnake, determine the equivalence factors
- Equivalence calculation run at each state-point to allow interpolation during the transient

Transient
- Run the eigenvalue calculation (to start with null-transient)
- Run the fully-coupled transient in Mammoth

Serpent



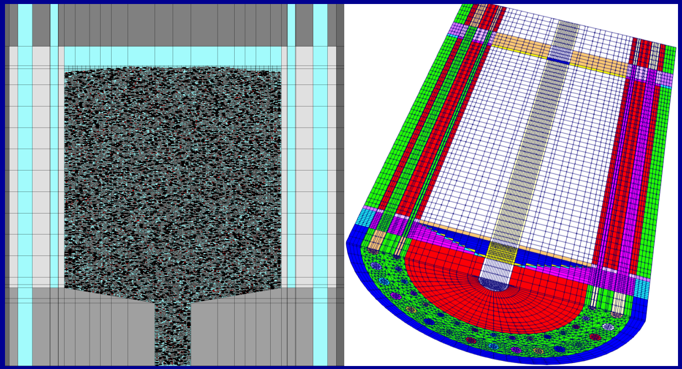


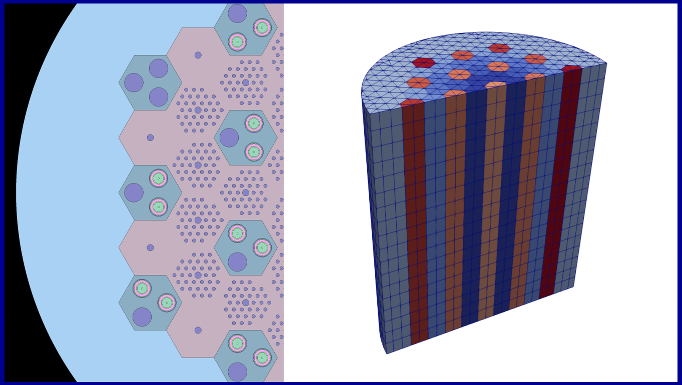
MAMMOTH
Diffusion w SPH

Advanced Equivalence Methods

- **HTR-PM**
  - Pebble bed HTR (420k pebbles)
  - Diffusion problem size – 54,656 cells

- **HTTR**
  - Prismatic HTR
  - Diffusion problem size – 15,552 cells





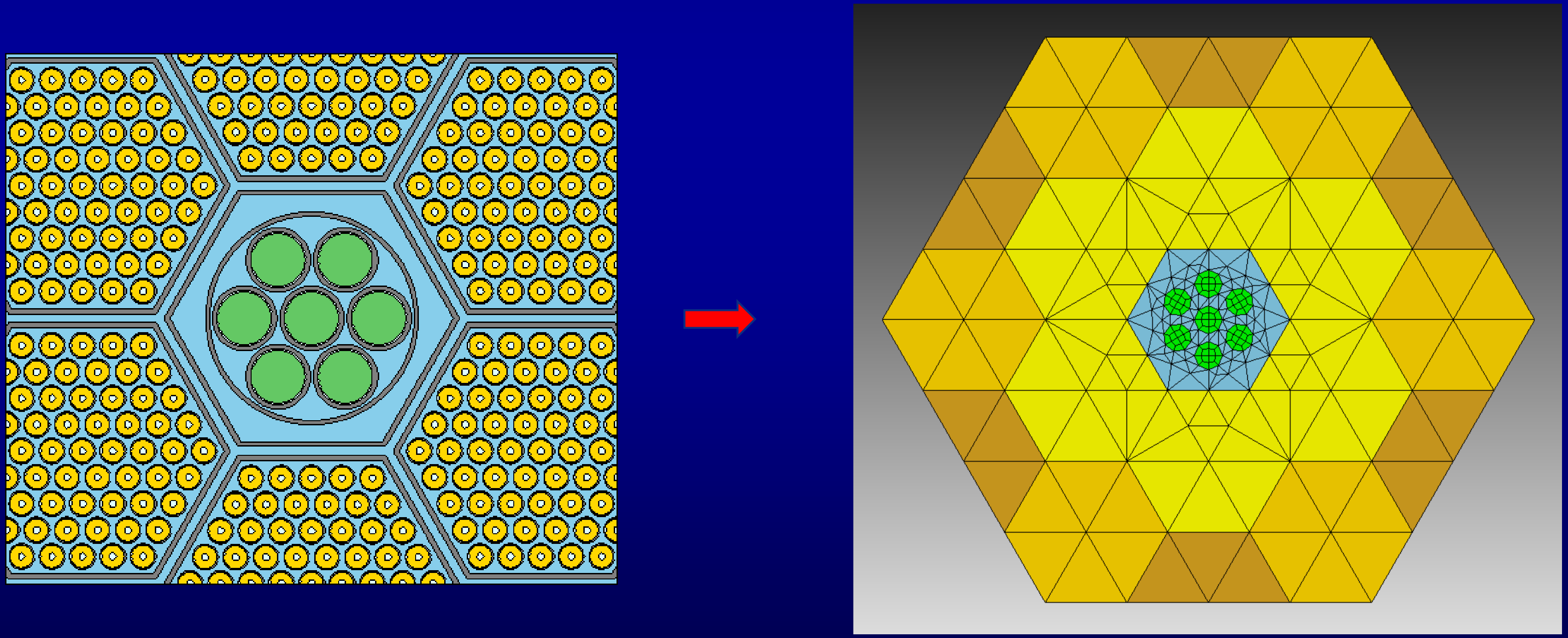
HTR-PM (T_{fuel} = 1100 K)	k_{eff}	Δpcm	RMS % Err $\nu\Sigma_f$	Max %Err $\nu\Sigma_f$
Serpent	1.01159	±1.8	-	-
Diffusion	1.03653	2435	6.0	40.6
SPH-Diffusion	1.01159	0	1.55E-03	2.94E-03

HTTR (T_{fuel} = 1300 K)	k_{eff}	Δpcm	RMS % Err Power	Max %Err Power
Serpent	1.00259	±2.7	-	-
Diffusion	1.01978	1715	3.12	6.20
SPH-Diffusion	1.00259	0	7.0E-02	2.0E-01

Transport level accuracy for the price of a diffusion calculation

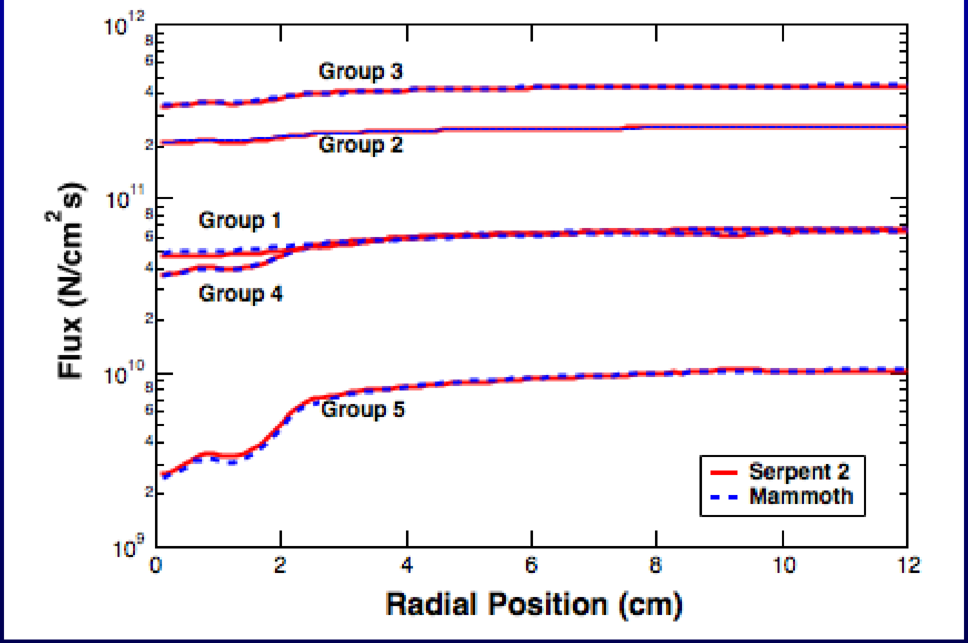
Multi-Scheme Capability

- MAMMOTH
  - Allows usage of transport where more detail is needed with efficiency of diffusion for remainder of domain



Multi-Scheme Capability

- MAMMOTH Results
  - Radial flux profile vs Serpent 2
    - Serpent results for bins having width of control rod
    - Mammoth results for a line

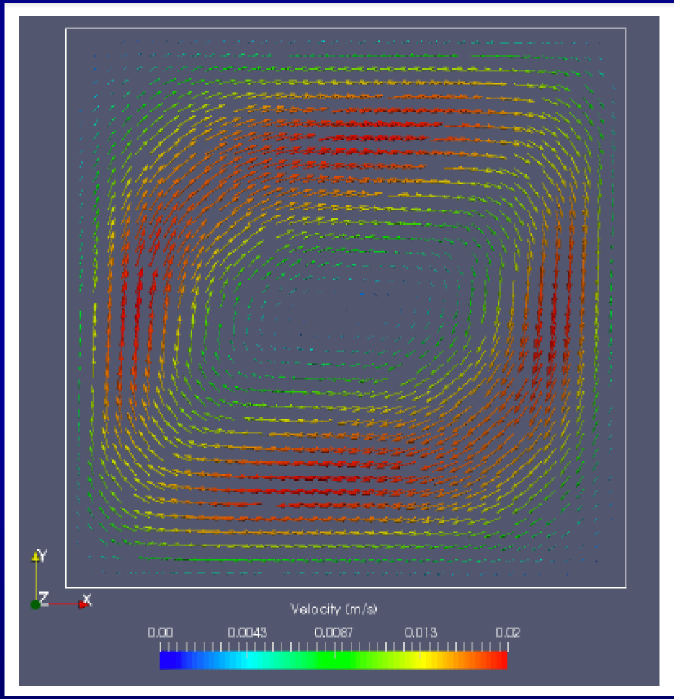


SAM: 3-D Flow Model

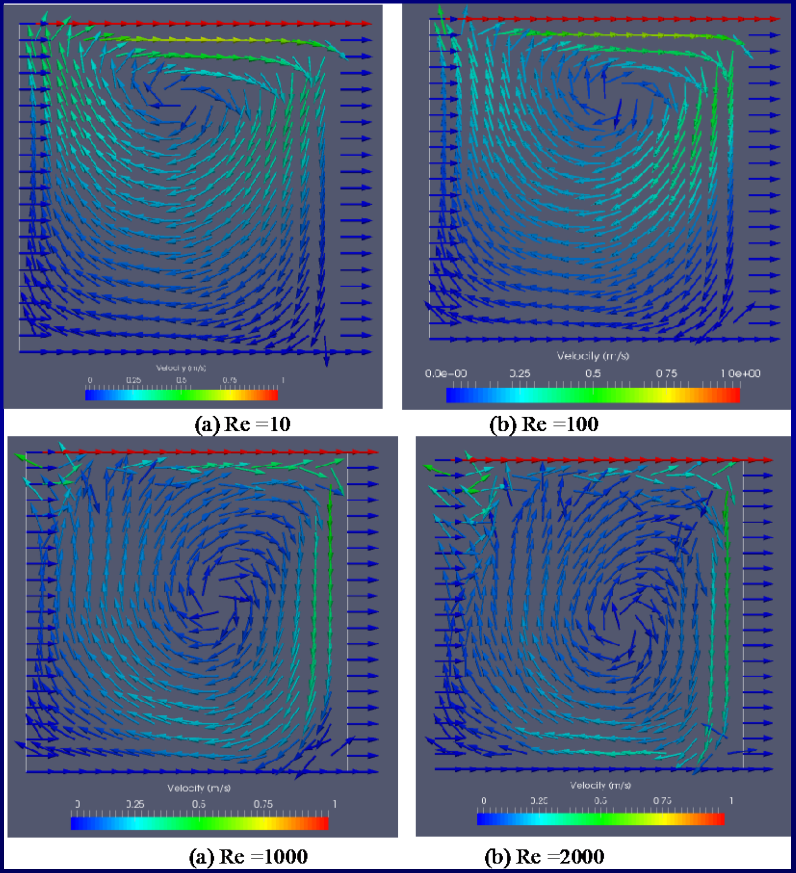
- SAM (Systems Analysis Module) is 2 codes in 1:
  - Component based system analysis code
    - Generates its own mesh
    - Has 1-D and 0-D components
  - 3-D unstructured mesh FEA capability (under development)
    - "Reduced Order" CFD-type flow solution for pools and porous media
      - (e.g.) molten salt fueled fast reactor (Terrapower)
    - 3-D solids with embedded 1-D flow channels
      - (e.g.) molten salt fueled thermal reactor (Terrestrial)
  - 1-D components and 3-D mesh are run concurrently using the MOOSE multi-app system

SAM: 3-D Flow Model

- Validation Examples



Natural Convection
Ra = 10^5



Lid-Driven Cavity Flow

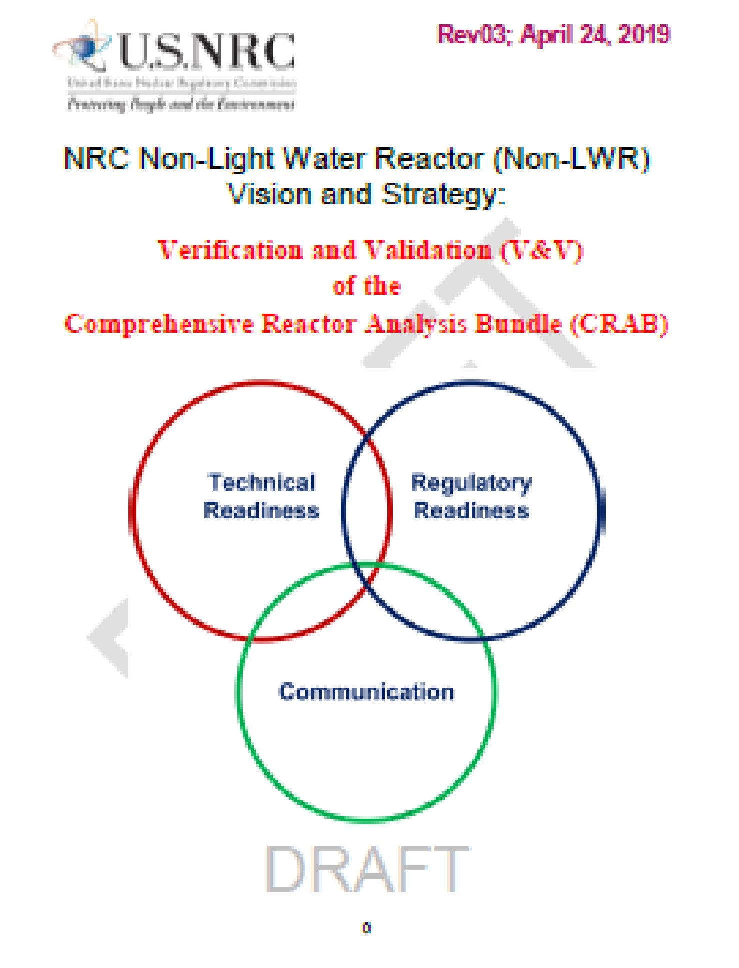
Verification & Validation “Gaps”

Code Assessment Issues

- Code “Assessment” = Verification & Validation represents the most significant “gap” in readiness for the DBE analysis codes.
- Verification: Considered generally good – however “coupling” may need additional cases to ensure conservation of mass, energy & momentum.
- Validation: Completed and on-going validation shows good agreement between predicted & measured results. More is needed, and should be done with a “frozen” code.

Code Validation Matrix

- Volume 1 identifies the most important validation cases for each of the 10 design types. Additional validation is being performed by DOE as part of developmental assessment.
- An additional report is being developed to summarize all of the V&V needed for CRAB.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Rev03; April 24, 2019

NRC Non-Light Water Reactor (Non-LWR)
Vision and Strategy:

Verification and Validation (V&V)
of the
Comprehensive Reactor Analysis Bundle (CRAB)

DRAFT

0

Approach to Validation for DBEs

(1) Review PIRT phenomena & prior test programs for applicability to each of the new designs.

(2) Identify and prioritize validation tests (based on PIRT findings and NRO expected review schedule).

(3) Develop "reference plant" models to define nodalization scheme and modeling options.

(4) Coordinate efforts with DOE and national labs to complete validation & improve code performance based on findings.

Validation Status

GCRs: HTR-10, PBMR-268,-400, SANA, HTTU, AVR, . . .

SFRs: EBR-II, FFTF, CEFR, ZPPR, Monju, . . .

LMRs: Helios

HPRs: KRUSTY, Godiva

MSRs: MSRE, UCB-Ciet, UW-Loop, . . .

RCCS: NSTF, UW-Loop, . . .

Completed
In-progress
Planned

- There are significant "gaps" : Validation is partial, with numerous tests in-progress or planned.
- More importantly, there is a lack of experimental data for some designs.

Validation / Experimental "Gaps"

GCRs: Prismatic gas-cooled IET (i.e. HTTR, OSU-HTTF)

SFRs: Pool type IET data, International data

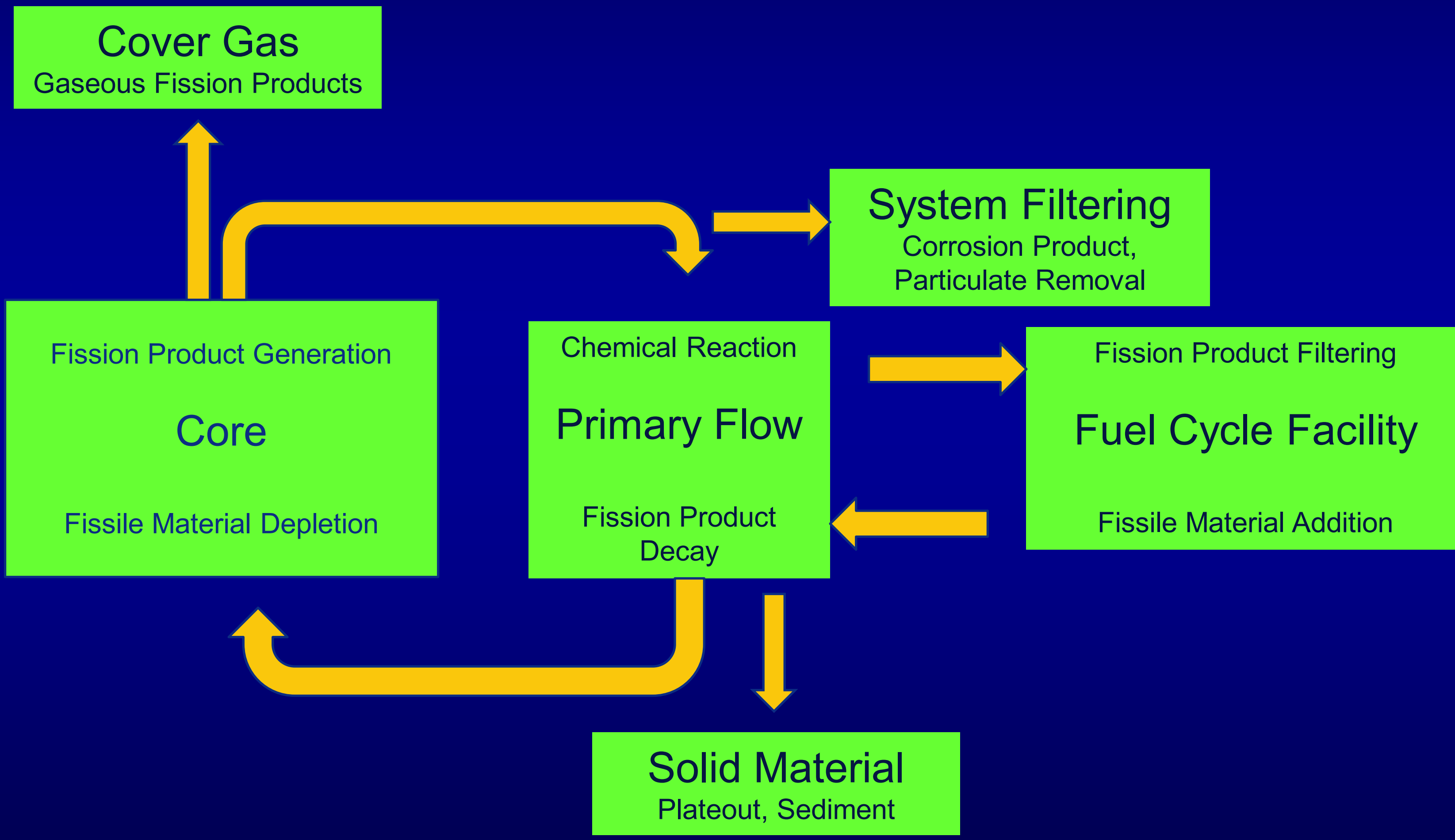
LMRs: Additional IET data, SET data for T/H, fuel, kinetics

HPRs: Monolith conduction and heat release SET data

MSRs: Pool type IET data, natural circulation loop data

Scaling of IETs and Range of Conditions of existing data to full-scale prototypes remains to be established.

Molten Salt Reactor (Inventory Control "Gap")



Code Readiness: Tracking "Gaps"

- Using PCMM (Predictive Capability Maturity Model) to characterize code readiness.
  - Geometric Fidelity
  - Physics and Model Fidelity
  - Code Verification
  - Solution Verification
  - Code Validation
  - Uncertainty Quantification
- Mauturity levels "0" to "3"

  (Think of it as "D" to "A")

Table 2: Predictive Capability Maturity Model (PCMM) Matrix

Element \ Maturity	Maturity Level 0	Maturity Level 1	Maturity Level 2	Maturity Level 3
Representation and Geometric Fidelity What features are neglected because of simplifications?	• Judgement only • Little or no representation or geometric fidelity for the system	• Significant simplification of the system	• Limited simplification of major components • Geometry is well defined for major components and some minor components • Some peer review conducted	• Essentially no simplifications made • Geometry of all components represented "as built" • Independent peer review conducted
Physics and Model Fidelity How fundamental are the physics and calibration of the models?	• Judgement only • Model forms are unknown or ad hoc • Few physics informed models • No coupling of models	• Some models and correlations are physics based and calibrated to data • Minimal or ad hoc coupling of models	• Physics based models and correlations for all important processes • Significant calibration using SETs and IETs • Some peer review conducted	• All models and correlations are physics based • Sound physical basis for extrapolation • Full coupling of models • Independent peer review conducted
Code Verification Are software errors and poor quality assurance practices?	• Judgement only • Minimum testing of software elements • Little or no SQA	• Code is managed by SQA procedures • Unit and regression testing performed	• Some algorithms are tested to determine convergence • Some features are tested with benchmark solutions • Some peer review conducted	• All of the important algorithms tested to determine convergence • All features and capabilities tested with rigorous benchmark solutions • Independent peer review conducted
Solution Verification Are numerical errors corrupting the results?	• Judgement only • Numerical errors are unknown or have large effect on results	• Numerical effects are qualitatively estimated • Input/output (I/O) verified only by analysis	• Numerical effects quantitatively estimated to be small • I/O independently verified • Some peer review conducted	• Numerical effects are determined to be small • Important simulations can be independently reproduced • Independent peer review conducted
Model Validation How carefully is the accuracy of the simulation and experimental results assessed?	• Judgement only • Few, if any comparisons to measurements in similar systems or applications	• Quantitative assessment of accuracy not directly relevant • Large or unknown experimental uncertainties	• Quantitative assessment of predictive accuracy for some key figures of merit from SETs and IETs • Experimental uncertainties well characterized • Some peer review conducted	• Quantitative assessment of predictive accuracy for all important figures of merit from SETs and IETs at conditions/geometries directly relevant to the application • Experimental uncertainties well characterized • Independent peer review conducted
Uncertainty Quantification and Sensitivity Analysis How thoroughly are uncertainties and sensitivities characterized?	• Judgement only • Only deterministic analyses conducted • Uncertainties and sensitivities not addressed.	• Aleatory and epistemic (A&E) uncertainties propagated, but without distinction • Informal sensitivity studies only	• A&E uncertainties propagated and identified • Quantitative sensitivity analyses conducted • Numerical propagation errors are estimated • Some strong assumptions made • Some peer review conducted	• A&E uncertainties comprehensively treated and properly interpreted • Comprehensive sensitivity analyses conducted • Numerical propagation demonstrated to be small • No significant assumptions • Independent peer review conducted

25

Summary & Conclusions

- **"Volume 1" recommends the codes in CRAB as the approach for non-LWR DBE analysis.**
- **"Gaps" in code capability, V&V are identified along with tasks for resolution.**
- **Using the combination of NRC and DOE codes will provide a <u>technically superior product</u> than can be attained with further development of only the NRC's conventional LWR codes.**

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Non-LWR SEVERE ACCIDENT AND SOURCE TERM RESEARCH ACTIVITIES & MELCOR DEVELOPMENT

Presented by:
Hossein Esmaili
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

May 1, 2019

Design Basis Source Term Development Process

(example: MOX & High Burnup Fuel)

Experimental Basis

PIRT process

- Oxidation/Gas Generation
- Melt Progression
- Fission Product Release
- Fission Product Transport

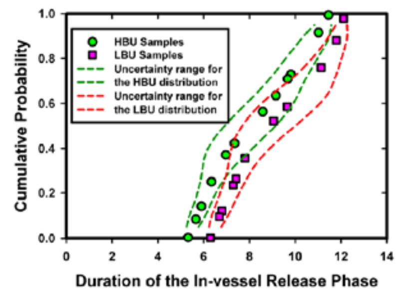
MELCOR

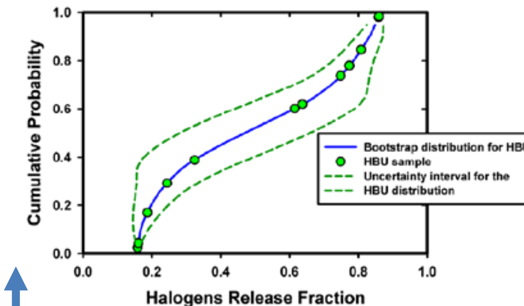
Accident Analysis

Scenario # 1	Scenario # 2
………………	………………
Scenario # n-1	Scenario # n

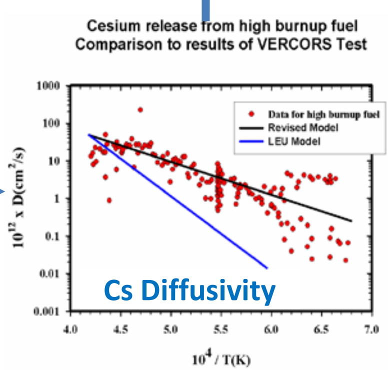
Synthesize timings and release fractions

Design Basis Source Term





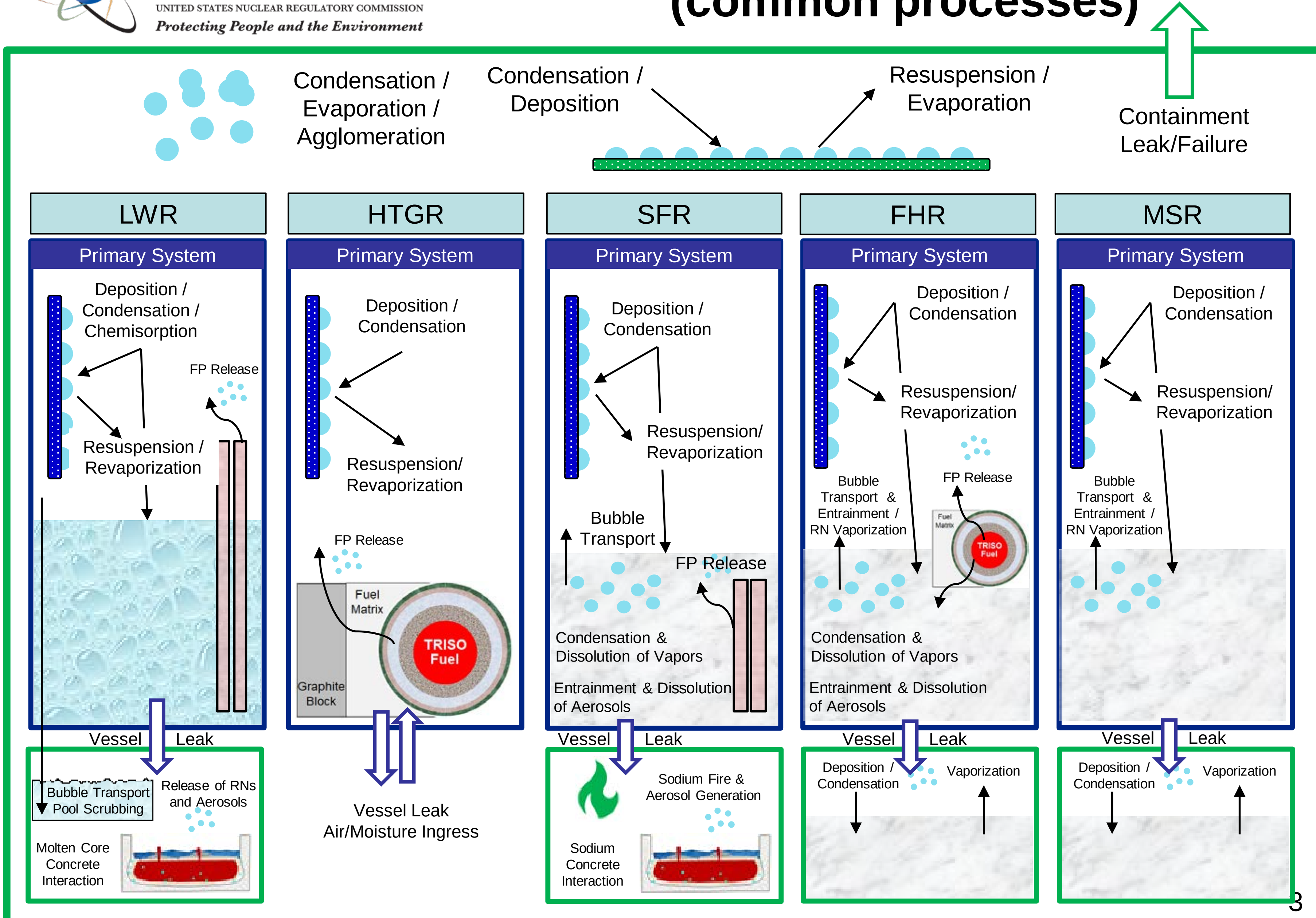
Fission Product class	Gap 0.5 hr	In-vessel 1.3 hr	Ex-vessel 2 hr	Late 10 hr
Noble Gases	5%	95%	~0	~0
Iodine, bromine	5%	35%	25%	1%
Cesium	5%	25%	35%	1%
Tellurium	~0	5%	25%	0.5%
Ba, Sr	~0	2%	10%	~0
Ru, Mo, Pd, etc.	~0	0.25%	0.25%	~0
Lanthanides	~0	0.02%	0.5%	~0
Cerium group	~0	0.05%	0.5%	~0



D. Powers, et al. "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup or MOX Fuel", SAND2011-0128 January 2011

- Similar RFs to NUREG-1465 but prolonged release
- Differences not from change of fuel but from code advances

Phenomenology & Release Paths (common processes)



Phenomena Identification and Knowledge Gaps

HTGR/FHR
Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), NUREG/CR-6944 , Volume 3: Fission-Product Transport and Dose PIRTs, 2008
Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), NUREG/CR-6944, Volume 2: Accident and Thermal Fluids Analysis PIRTs, 2008
Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), NUREG/CR-6944, Volume 5: Graphite PIRTs, 2008
TRISO-Coated Particle Fuel Phenomena Identification and Ranking Tables (PIRTs) for Fission-Product Transport Due to Manufacturing, Operations, and Accidents, NUREG/CR-6844, 2004

SFR
Advanced Sodium Fast Reactor Accident Source Terms: Research Needs, SAND2010-5506, Sandia National Laboratories, 2010.
Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety, SAND2011-4145, Sandia National Laboratories, 2011
Regulatory Technology Plan: Sodium Fast Reactor: Mechanistic Source Term Development," Argonne National Laboratories, ANL-ART-3, 2015
Regulatory Technology Development Plan: Sodium Fast Reactor : Mechanistic Source Term - Metal Fuel Radionuclide Release, Argonne National Laboratories, ANL-ART-38, 2016

MSR/FHR
Planned: Canadian Nuclear Laboratories, Chalk River (Summer 2019)
Phenomena Important in Modeling and Simulation of Molten Salt Reactors (BNL-114869-2018-IR, 2018)
Phenomena, gaps, and issues for neutronics modeling and simulation of FHRs (Annals of Nuclear Energy, 2019)

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HTGR Development

Key Phenomenon	Importance	Existing Capabilities	Modeling Gaps
Modeling of TRISO fuels	Determining release of fission products from fuel and fuel material properties	• Analytic release model • Multi-zone diffusion model • Account for FP recoil, matrix contamination, and initial TRISO defects	• Current modeling uses UO2 material properties, needs to be extended to UCO (Development Items M2.1 and M2.2)
Heat Transfer in Graphite block (PMR)	Thermal response of fuel components and failure of TRISO fuel particles	• Tanaka-Chisaka effective radial conductivity	
Heat Transfer in fuel pebbles (PBR)	Thermal response of fuel components and failure of TRISO fuel particles	• Zehner-Schlunder-Bauer effective thermal conduction	
Reactivity temperature feedback coefficients.	Neutronics power feedback	• Point kinetics model • Reactivity coefficients specific to an application can be implemented via control functions	
Ability to model two-sided reflector component	Heat transfer from overheated core	• Two-sided reflector component	
Modeling graphite dust transport	Pathway for fission product transport and release	• All relevant mechanisms for graphite dust transport, deposition, and resuspension	
Graphite oxidation	Heat generation and release of combustible gases	• Graphite oxidation model and oxidation products	
Air/moisture Ingress modeling	Air/moisture ingress can lead to oxidation of the graphite structures and release of radionuclides	• Momentum exchange model	

SFR Development

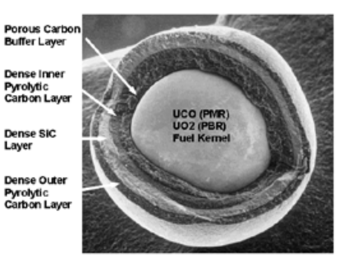
Key Phenomenon	Importance	Existing Capabilities	Modeling Gaps
Liquid Metal as Working Fluid	Modeling the liquid metal coolant heat transfer properties is essential in simulating the reactor response to accident conditions	Na equation of state (EOS) libraries already available to MELCOR.	• Ability to model sodium as the working fluid in some control volumes and water in others will be added (development Item M1.7) • Addition of Pb and Pb/Bi EOS/Properties (infrastructure developed under development item M1.7)
Sodium Fire & Sodium Concrete Interaction Modeling	Fire is a source of heat and provides a path for transport of sodium and fission products to the atmosphere. Concrete interaction important source of aerosols and possible combustible gases.	Sodium pool fire and spray fire models, as well as atmospheric chemistry models have already been added to the code.	• Addition of a hot gas layer model during sodium fires (development Item M1.6) • Add sodium concrete interactions (development Item M1.5)
Fission Product Speciation	Affects the release, vapor pressure, and chemical interactions of fission products.	MELCOR radionuclide classes organized by chemical similarities can be easily adapted for reactor application	• Determination of MELCOR class structures (development Item M1.3)
Fission Product Release Model	Determines distribution of fission products between the fuel and fission gas plenum.	MELCOR has a generic release model easily adapted for metallic fuel.	• Extension of existing modeling for FP release for metallic fuel (development Item M1.4)
Fuel Degradation Model	Degraded fuel components lead to release of fission products from the fission gas plenum as well as some fuel/clad material.	MELCOR has models for fuel components that can be extended to SFR application	• Extend MELCOR fuel component to capture melting fuel in fuel matrix • Model for cladding failure from eutectic penetration or molten fuel contact • Ejection of fuel/sodium from failed rod (Item M1.2)
Dissolution of RN and vaporization of dissolved species	Transport of radionuclides to and from the sodium pool and into the cover gas	Similar capability exists for molten corium pool (VANESA)	• Add models for dissolution and vaporization of dissolved species (development Item M1.3)
Bubble Transport/ partitioning between bubble & sodium pool	Transport of radionuclides directly to the atmosphere.	MELCOR's SPARC model might be leveraged, though modified significantly for this application	• Development of bubble transport model (development Item M1.3)
Heat Pipe Thermal Hydraulics & Failure	The heat pipe is the primary means of heat removal from fuel. Failure of the heat pipe determines the extent of core degradation and source term released from fuel.	Existing multi-rod model can be leveraged in calculating propagation of local heat pipe failure (development Item 1.8)	• MELCOR does not currently have a heat pipe model. Code modifications have been proposed to remove the gaps (development Item M1.1 & M1.8)
Reactor kinetics	Calculate transient power feedback	Existing point kinetics and reactivity feedback model	• Evaluate neutronics parameters in the existing point kinetics model (development Item M1.9)

MSR Development

Key Phenomenon	Importance	Existing Capabilities	Modeling Gaps
Physical Properties	Fundamental to simulation of steady state temperature and flow distributions.	FLiBe EOS and properties already implemented in MELCOR.	Validation of properties (development items M3.1, M3.4 and M3.6)
Heat Transfer Coefficients	Transfer of heat to calculate heat loads to structural materials	Existing generic correlation forms	Implement and validation of heat transfer coefficients (development items M3.4 and M3.6)
Track the flow of gas through the molten salt	Important for calculating entrainment of fission products from molten salt (next item)	SPARC model for aerosol scrubbing in liquid pools exists in MELCOR	Extend the SPARC model and bubble rise model (development items M3.2).
Entrainment of contaminated molten salt droplets in the gas flow	The primary mechanism for such entrainment of droplets is of course the rupture of gas bubbles at the molten salt surface.	Similar capability exists for molten corium pool (VANESA)	Use of correlations derived from data for droplet formation during bubble bursting in aqueous systems. This phenomenon is described further in Appendix C and is part of development Item M3.2 MSR
Vaporization of fission products from the molten salt.	Release of volatile fission products to cover gas.	Similar capability exists for molten corium pool (VANESA)	This phenomenon is described further in Appendix C and is part of Development Items M3.2 and M3.3 MSR

Input & Data Requirements

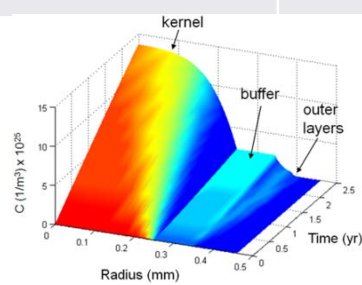
Input Data	HTGR	SFR	MSR	FHR
FP Inventory	SCALE	SCALE	SCALE	SCALE
FP diffusion coefficients (D) and release	Experiments (e.g., AGR) and analysis (e.g., DOE tools)	Experiments		Experiments (e.g., AGR) and analysis (e.g., DOE tools)
Core power shape	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (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)



SCALE

$$\frac{\partial C}{\partial t} = \frac{1}{r^m}\frac{\partial}{\partial r}\left(r^m \boxed{D}\frac{\partial C}{\partial r}\right) - \lambda C + \boxed{S}$$

Experiments/Analysis

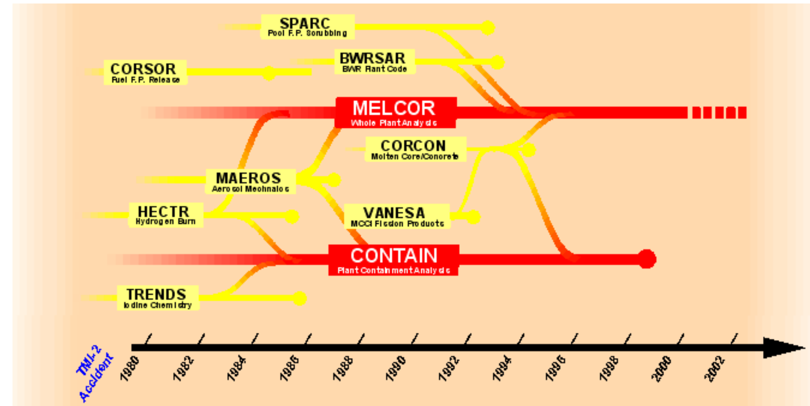


Distribution of Cs-137 in different layers as a function of time

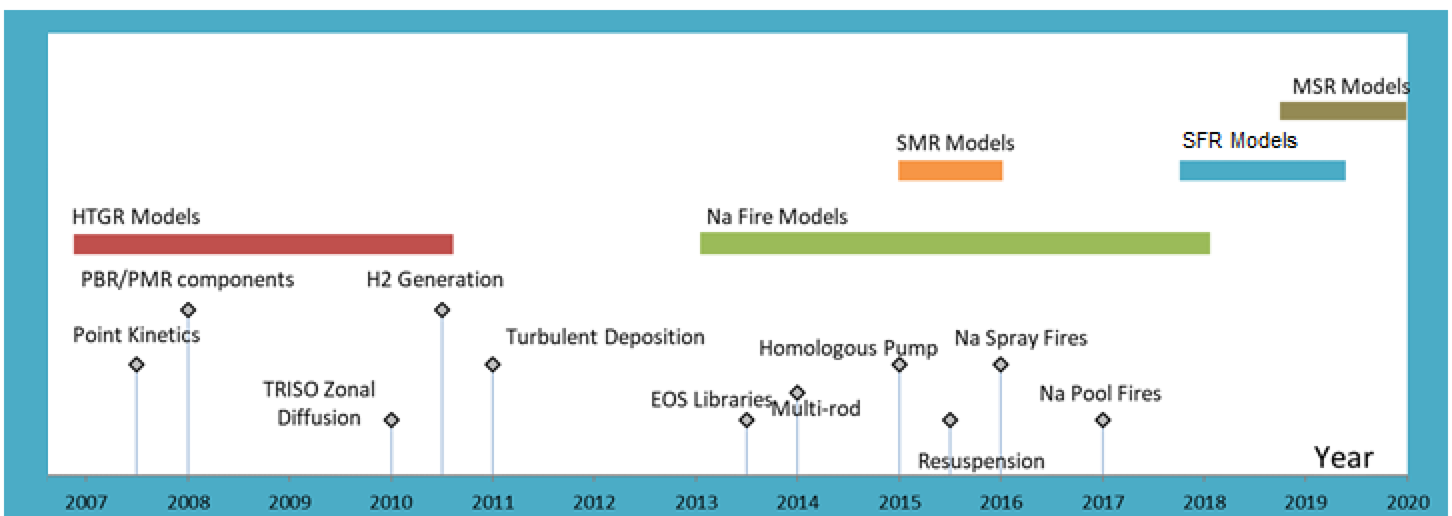
MELCOR Integrated Analysis

MELCOR developed at Sandia National Laboratories for the U.S. NRC

- State-of-the-art tool for severe accident progression and source term analysis. Ongoing development of new capabilities including models for non-LWR applications
- Replace collection of simple, special purpose codes, i.e., Source Term Code Package (STCP)
- Eliminate tedious hand-coupling between modules
- Capture feedback effects (i.e., coupling of temperatures, release rates, and decay heating)



Other integrated severe accident tools (e.g., MAAP or ASTEC) do not have non-LWR capabilities of MELCOR



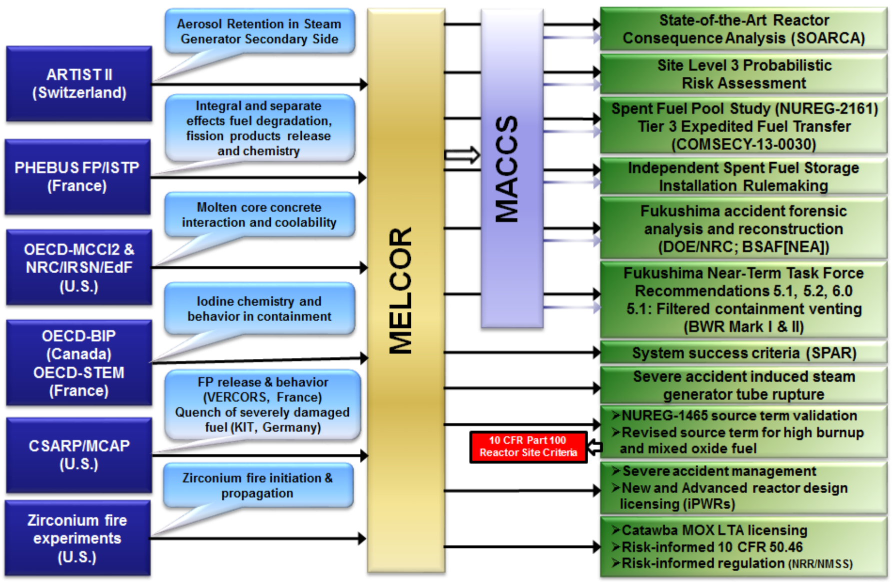
Beyond FY20, code development activities will focus on specific advanced reactor technologies and design specific modifications and code assessments as those details and funding become available.

FY202x

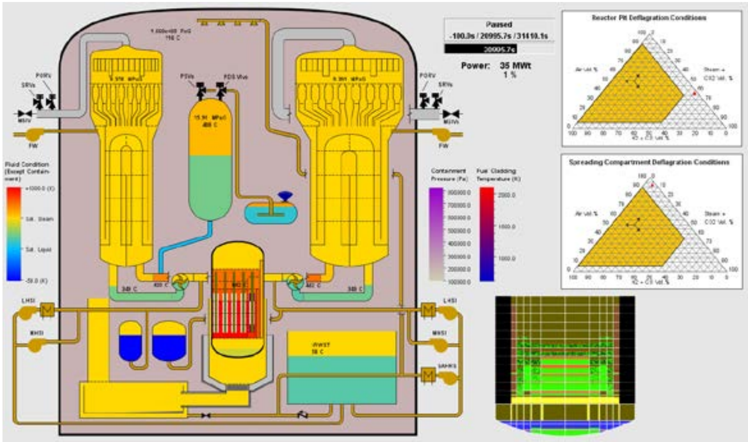
MELCOR Code Development

- Fully Integrated, engineering-level code
  - Thermal-hydraulic response in the reactor coolant system, reactor cavity, and containment;
  - Core heat-up, degradation, and relocation;
  - Core-concrete interaction;
  - Hydrogen production, transport, and combustion;
  - Fission product release and transport behavior
- Traditional Application
  - User constructs models from basic constructs
    - Control volumes, flow paths, heat structures,
  - Multiple 'CORE' designs
    - PWR, BWR, HTGR (Pebble Bed & PMR), PWR-SFP, BWR-SFP, SMR, Sodium (Containment)
  - Adaptability to new reactor designs
- Validated physical models
  - ISPs, benchmarks, experiments, accidents
- Probabilistic Risk Analysis (e.g., Level 3 PRA)
- Uncertainty Analysis (e.g., SOARCA)
  - Relatively fast-running
  - Characterized numerical variance
- User Convenience
  - Windows/Linux versions
  - Utilities for constructing input decks (GUI)
  - Capabilities for post-processing, visualization
  - Extensive documentation
- Small Modular Reactors
  - NuScale
- Non-LWR Reactors
  - HTGR/SFR/MSR

Code Development & Regulatory Applications



International Collaboration (CSARP/MCAP/EMUG/AMUG)



Integrated models required for self-consistent analysis

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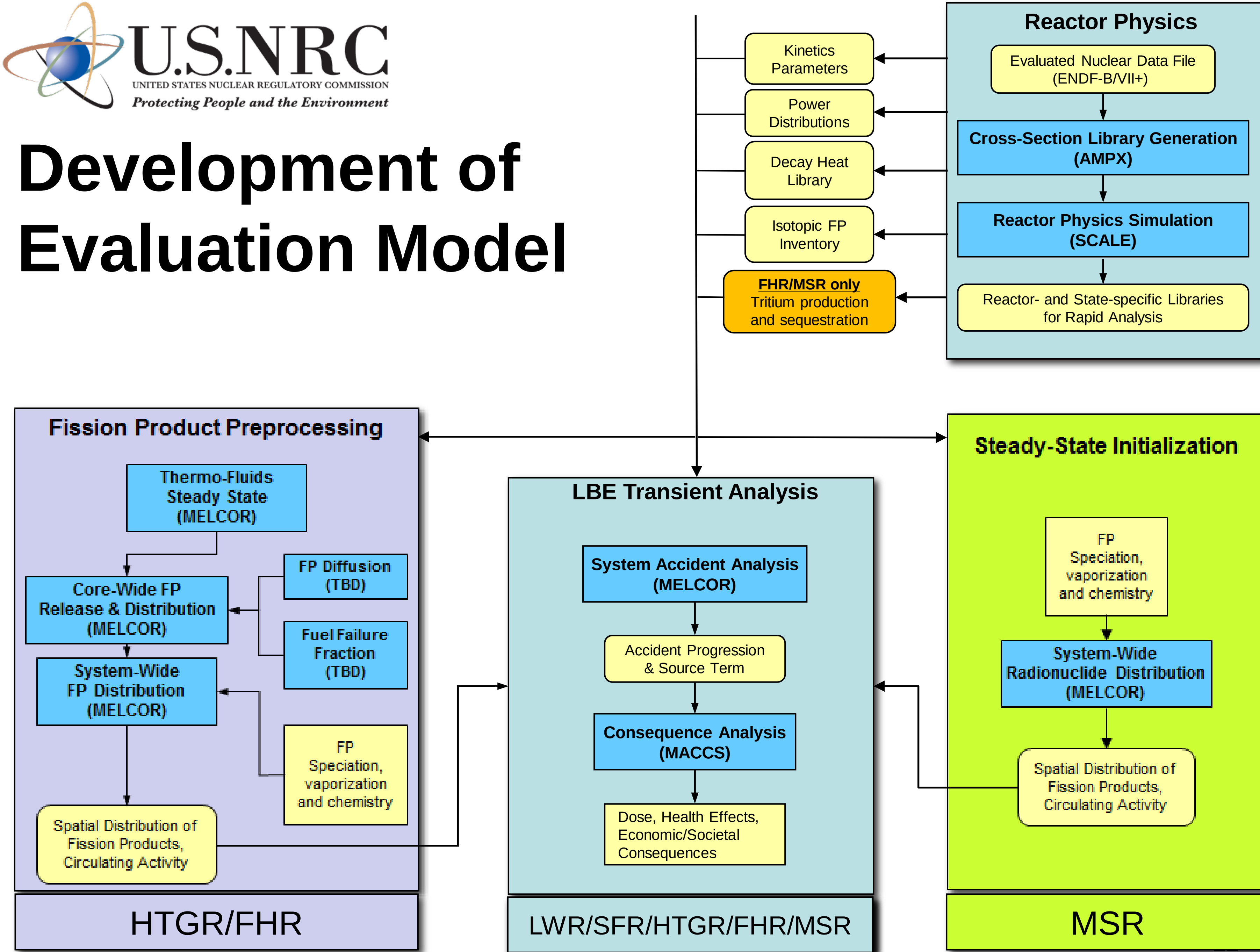
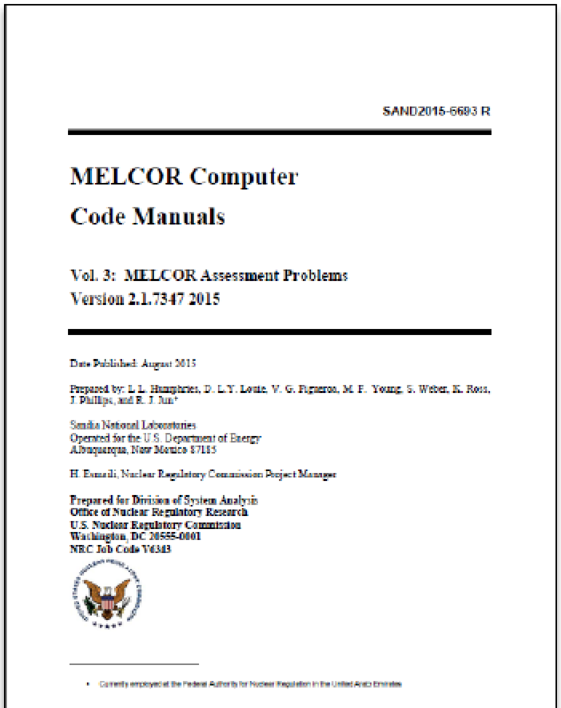
Development of Evaluation Model



Verification & Validation of Models



SAND2015-6693 R

MELCOR Computer
Code Manuals

Vol. 3: MELCOR Assessment Problems
Version 2.1.7347 2015

Date Published: August 2015

Prepared by: L. L. Humphries, D. L.Y. Louie, V. G. Figueroa, M. F. Young, S. Weber, K. Ross, J. Phillips, and R. J. Jun*

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Prepared for Division of System Analysis
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U.S. Nuclear Regulatory Commission
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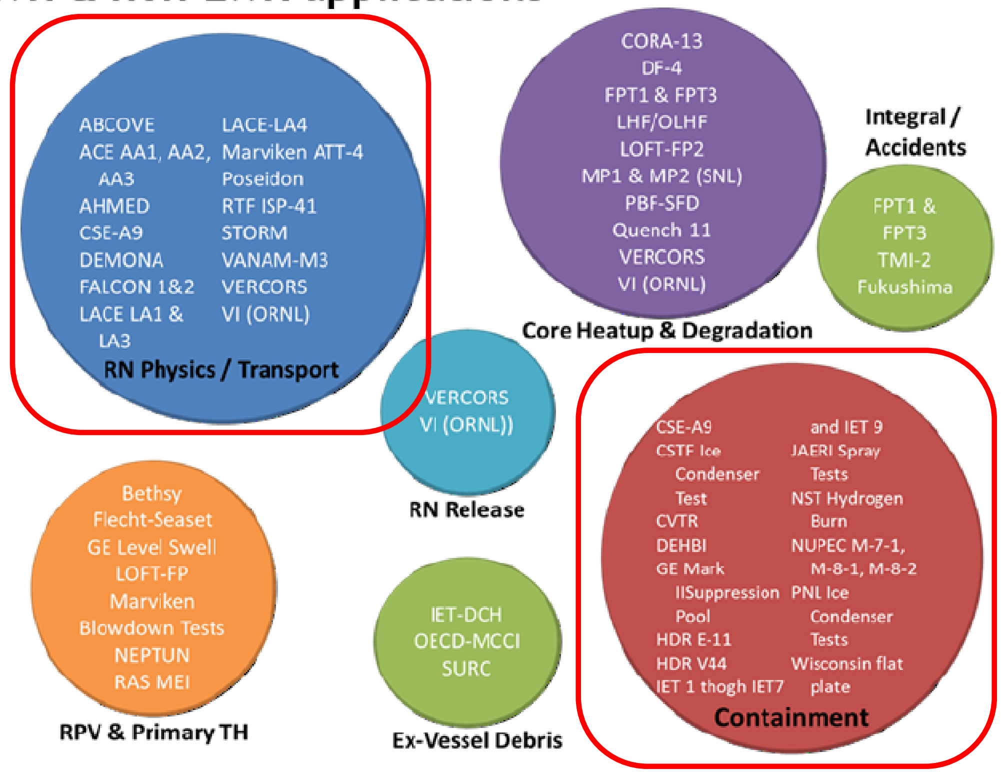
NRC Job Code V6343

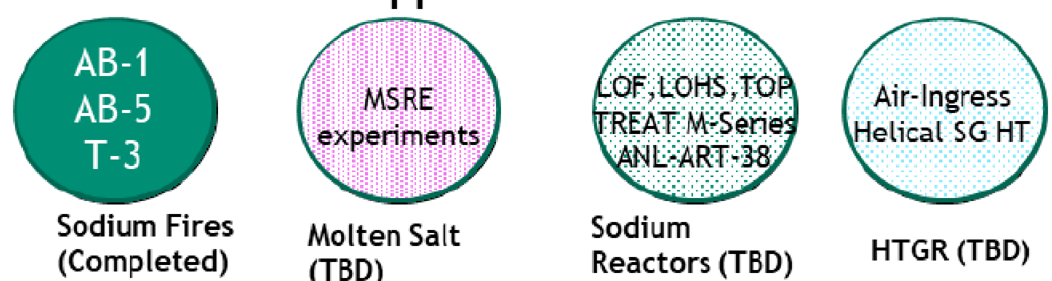
Volume III: Assessments

SAND2015-6693 R

LWR & non-LWR applications



Specific to non-LWR application



Concluding Remarks

- Leveraging decades of model development and validation that can be extended to non-LWRs
  - Most efficient approach for accident progression and source term analysis
  - For some technologies, the models are ready to be tested
- Plan will be updated as more experience is gained and as new information regarding specific reactor designs becomes available

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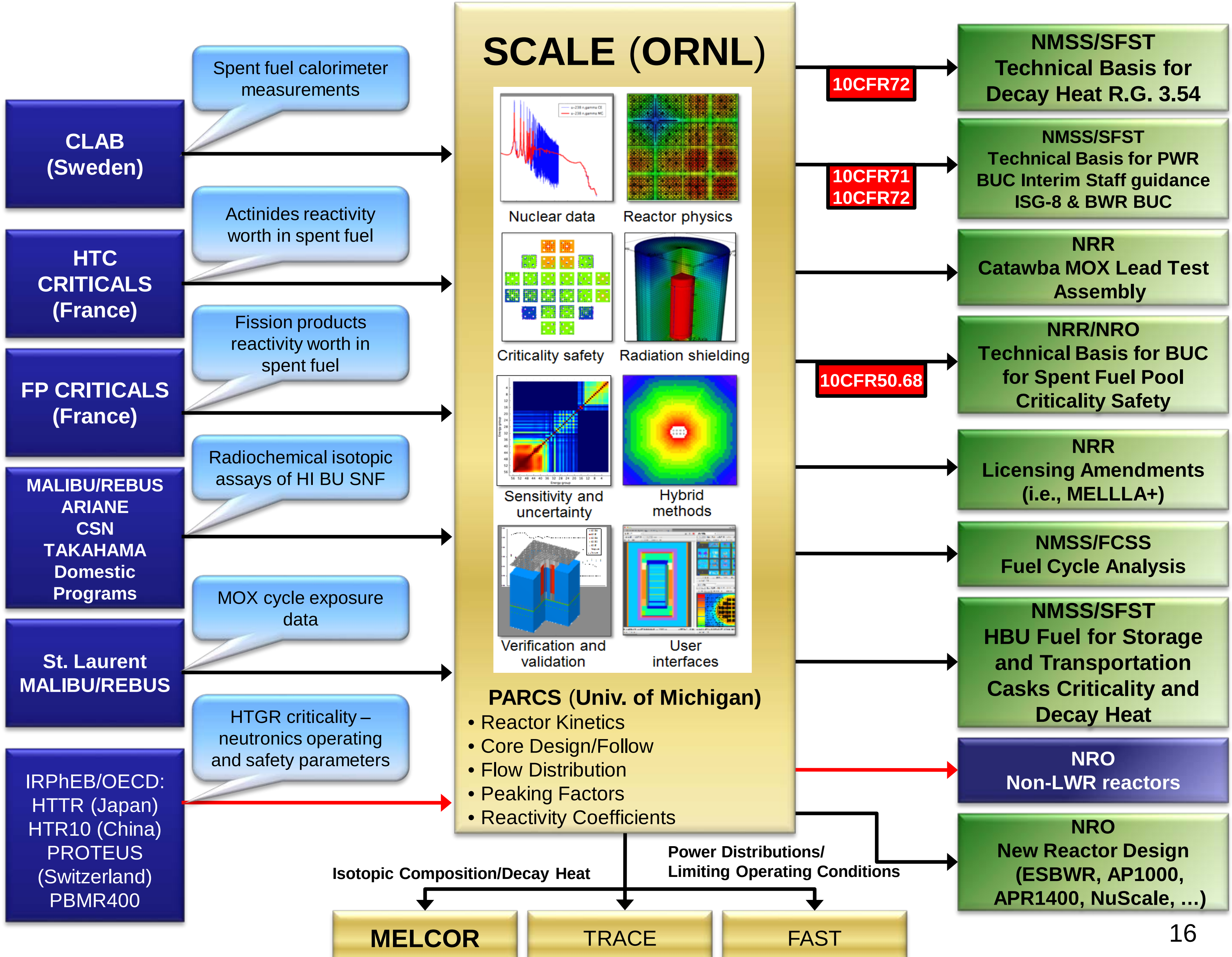
SCALE Non-LWR ACTIVITIES

Don R. Algama
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

May 1, 2019

General Approach

- Overview of phenomena important to licensing for Accident Progression
- What is the validation basis for these phenomena,
  - highlight gaps
  - is a bounding analysis acceptable?
- Build upon existing tools, nuclear data and validation


SCALE (ORNL)
Nuclear data
Reactor physics
Criticality safety
Radiation shielding
Sensitivity and uncertainty
Hybrid methods
Verification and validation
User interfaces
PARCS (Univ. of Michigan)
• Reactor Kinetics
• Core Design/Follow
• Flow Distribution
• Peaking Factors
• Reactivity Coefficients
CLAB (Sweden)
Spent fuel calorimeter measurements
HTC CRITICALS (France)
Actinides reactivity worth in spent fuel
FP CRITICALS (France)
Fission products reactivity worth in spent fuel
MALIBU/REBUS ARIANE CSN TAKAHAMA Domestic Programs
Radiochemical isotopic assays of HI BU SNF
St. Laurent MALIBU/REBUS
MOX cycle exposure data
IRPhEB/OECD: HTTR (Japan) HTR10 (China) PROTEUS (Switzerland) PBMR400
HTGR criticality – neutronics operating and safety parameters
10CFR72
NMSS/SFST Technical Basis for Decay Heat R.G. 3.54
10CFR71 10CFR72
NMSS/SFST Technical Basis for PWR BUC Interim Staff guidance ISG-8 & BWR BUC
NRR Catawba MOX Lead Test Assembly
10CFR50.68
NRR/NRO Technical Basis for BUC for Spent Fuel Pool Criticality Safety
NRR Licensing Amendments (i.e., MELLLA+)
NMSS/FCSS Fuel Cycle Analysis
NMSS/SFST HBU Fuel for Storage and Transportation Casks Criticality and Decay Heat
NRO Non-LWR reactors
NRO New Reactor Design (ESBWR, AP1000, APR1400, NuScale, …)
Isotopic Composition/Decay Heat
Power Distributions/ Limiting Operating Conditions
MELCOR
TRACE
FAST

SCALE for MELCOR/MACCS (LWRs)

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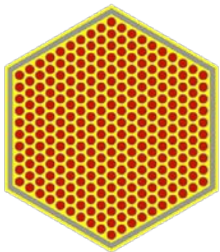
scale
Nuclear Systems Modeling & Simulation



ENDF/B
Physics data
Thermal scattering law, resonance data, energy distributions, fission yields, decay constants, etc.

AMPX
Validated cross section libraries; depletion and decay data

TRITON / Polaris
Transport and depletion in 1D, 2D, and 3D for LWR, ATF, and nonLWR

ORIGEN / ORIGAMI
Depletion, activation and decay
Reactor-specific radioactive source term characterization

Severe accident data generation with SCALE for **LWRs**

1. accumulate/curate nuclear data (years of data expert time)
2. develop methods (years of code developer time)
3. perform decay heat, isotopics validation (months of analyst time for a specific data set)
4. analyze operating history sensitivity for a specific reactor fuel, e.g. effect of control rods inserted during depletion (months of analyst time)
5. analyze sensitivity of a specific accident scenario analyze to modeling details, e.g. model axial spectral effect in fuel (weeks of analyst time)
6. generate isotopics at various decay times with uncertainty/bias envelopes (weeks of analyst time)

Non-LWR technology matrix

Reactor Type	Fuel / Enrich	Thermal spectrum	Fast Spectrum	Coolant	Radial core expansion	Flowing Fuel	Fuel Form	Control elements
HPR	~20%		✓	Sodium heat pipes	✓		Metallic Castings	External drums
	19.75%	Thermal/ Epithermal		Sodium heat pipes (dual condenser)			Oxide	External drums
SFR	~20%		✓	Sodium	✓		Metallic Rods	Internal rods
	~20%		✓	Sodium	✓		Metallic Rods	Internal rods
LFR	15-20%		✓	Lead	✓		Oxide/ Nitride	Internal rods
HTGR	15.5%	✓		Helium		Pebbles	TRISO	External rods
	~20%	✓		Helium			TRISO	Internal rods
FHR	~17%	✓		FLiBe		Pebbles	TRISO	External rods

Non-LWR technology matrix

Reactor Type	Fuel / Enrich	Thermal spectrum	Fast Spectrum	Coolant	Radial core expansion	Flowing Fuel	Fuel Form	Control elements
MSR	~5%	✓		Proprietary		Salt	Molten Salt	External rods
	~20%		✓	Chloride salt		Salt	Molten Salt	External rods?
	~20%		✓	Chloride salt		Salt	Molten Salt	
	Thorium	✓		FLiBe		Salt	Molten Salt	Internal rods

SCALE for MELCOR/MACCS (Non-LWRs)

- **Adapted design specific LWR approach as described in Section 3 of the Volume 3 BDBA plan**
- **Numerous SCALE calculations may be required to explore parameter space**
  - focus on bounding analyses with user assigned operating history (e.g. fuel temperature changes, reactor power changes)
  - understand bias and uncertainty through validation

HTGR/FHR Considerations:
- ~200,000 pebbles, each with ~8-10,000 randomly placed TRISO particles (UO2; UCO)
- Flowing pebbles; ~30 day residence; graphite blanks
- TRISO stationary in graphite blocks (Prismatic)
- Helium coolant
- Liquid salt (FLiBe) coolant, with salt requiring thermochemical analysis capability; filtration and salt flow rates

SFR/HPR Considerations:
- Solid fueled
- Reactivity effects due to thermal expansion
- Tightly coupled core; fast neutron leakage
- Salt voiding
- Salt thermochemical equilibrium state

MSR (liquid fuel) Considerations:
- Liquid fueled; neutron precursor drift
- Complicated chemical processes, filtration, feed systems;
- Fast flow rates of liquid fuel around primary circuit

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Protecting People and the Environment

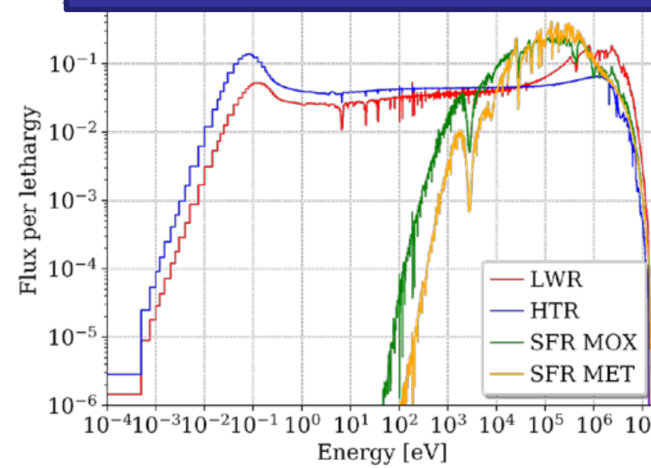
SCALE: Current Status

- **Nuclear Data (neutron; coupled neutron-gamma)**
  - Maximum fidelity CE and activation/decay nuclear data libraries (300,000 point wise data)
  - Lower fidelity (faster running) MG for LWR available/non-LWR in development (56grp, 252grp, 1597grp)
- **Validated methods for fuel evolution in LWRs**
  - Decay heat comparison to measurement
    - ~100 PWR and BWR comparisons
    - burst fission
  - Destructive assay
    - Isotopics from ~100 PWR and BWR cases
    - Assessments included detailed operating history
  - Criticality (ensures calculated neutron energy spectrum is reasonable)
    - Assessed against ~800 critical benchmarks
  - Shielding
    - Sparse data but looking at SINBAD database
- **Validation for fuel evolution in non-LWRs depends on**
  - declaration of reference designs and collection of relevant experiments (validation data)
  - **any methods changes should be based on validation results**

Changes in $^{35}Cl(n,p)$ cross section from ENDF/B-VII.0 to VII.1 induce ~2000 pcm reactivity change in Chloride salt MSR

Data Library	k_{eff}
ENDF/B-VII.0	1.02993
ENDF/B-VII.1	1.04924

Flux spectra in different systems



SCALE validation more info: https://www.ornl.gov/scale/validation

Next steps for SCALE for Non-LWRs

Nuclear Data and Libraries

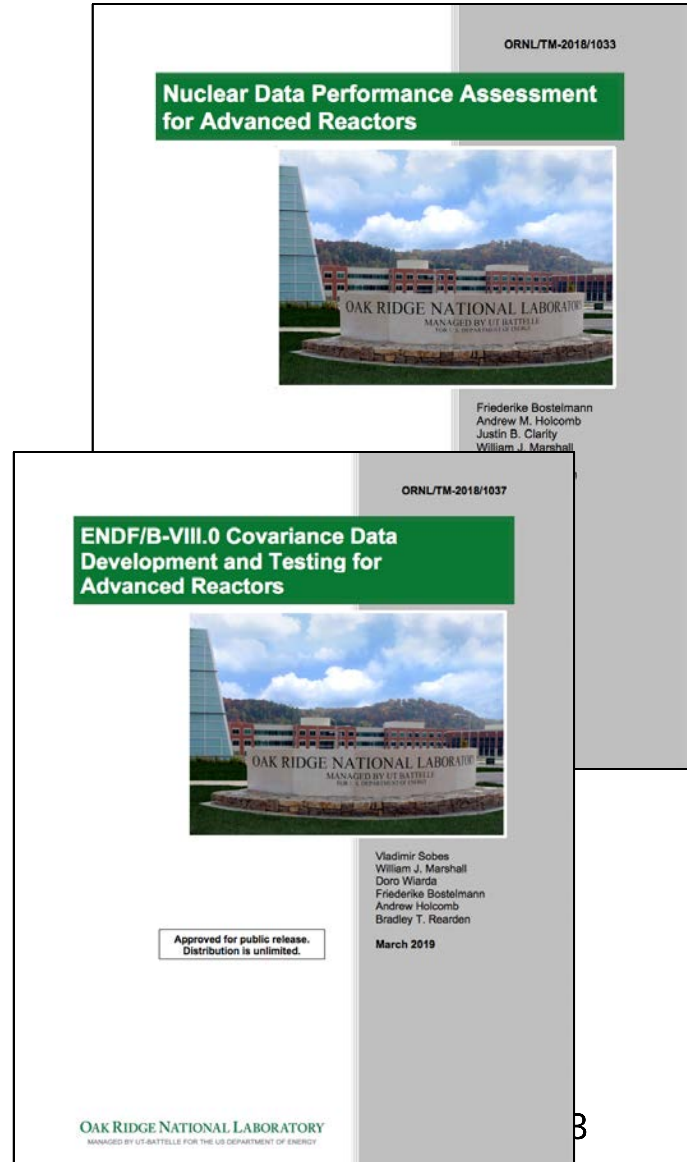
1. Assess ENDF libraries against Non-LWR designs
2. Assess system-specific multigroup libraries, especially for depletion and nuclide inventory predictions

Neutronics Capabilities

1. Assess Depletion
2. Assess Criticality
3. Assess Sensitivity/Uncertainty Analysis Methods

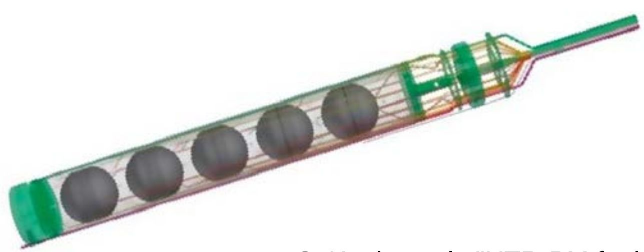
Data Gaps for Power and Flux Distributions

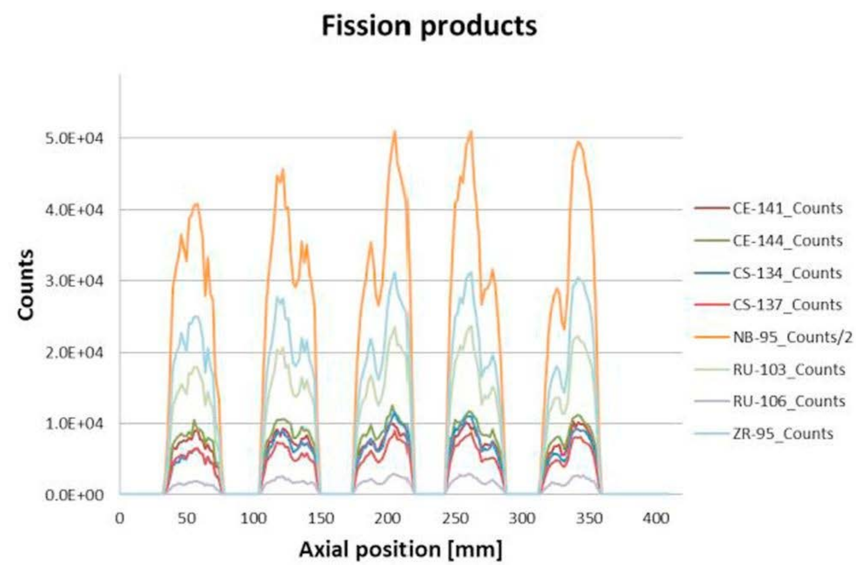
- Assessing data gaps in non-LWR requires
  - system specifics (composition+geometry/spectrum)
  - definition of accident scenarios and acceptable error
  - isotopics validation cases for similar systems
- Examples
  - HALEU validation with >5w/o
  - Thermal scattering law data, e.g. S(alpha,beta)
    - missing for some materials (FLiBe / FLiNaK) could cause >2000 pcm error in thermal systems
    - new ENDF/B-VIII graphite data is primarily based on simulations--needs more validation
  - U-233 criticality validation results poor (~2000 pcm)



Data Gaps for Depletion, Isotopics and Decay Heat

- non-LWR fuel isotopics and decay heat validation data is limited for
  - molten salt fuel and coolant evolution (e.g. MSRE tritium production data)
  - fast spectrum fuel evolution (EBR-II provides some initial data, VTR, etc)
  - TRISO-based fuel evolution (e.g. AGR-1 post-irradiation examination, HTR-PM qualification)
- criticality benchmarks gaps, e.g.:
  - High Temperature range
  - Reactor grade graphite
  - MSR (FLiBe, FLiNaK)





S. Knol, et. al., "HTR-PM fuel pebble irradiation qualification in the high flux reactor in Petten", Nuclear Engineering and Design, Volume 329, 2018.

Current Focus

- How data transfer will work between SCALE and MELCOR/MACCS
- Moving fuel and power history presents challenges
  - Demonstrate the sufficiency of bounding analysis for licensing use
- Correct level of chemistry modelling between SCALE and MELCOR

Summary

- Leveraging decades of physics models, nuclear data, and validation that can be extended to non-LWRs
  - Most efficient approach to support accident progression and source term analysis
  - For some technologies, the models are ready to be tested
- Experimental Needs
  - Decay heat, isotopic, validation data consistent with design and expected operating envelope
  - Criticality benchmarks
  - Destructive assay data for new fuel forms (e.g.: TRISO)
- Capabilities will be enhanced as more experience is gained, and gaps and uncertainties are quantified
- Plan will be updated as more experience is gained and as new information regarding specific reactor design becomes available

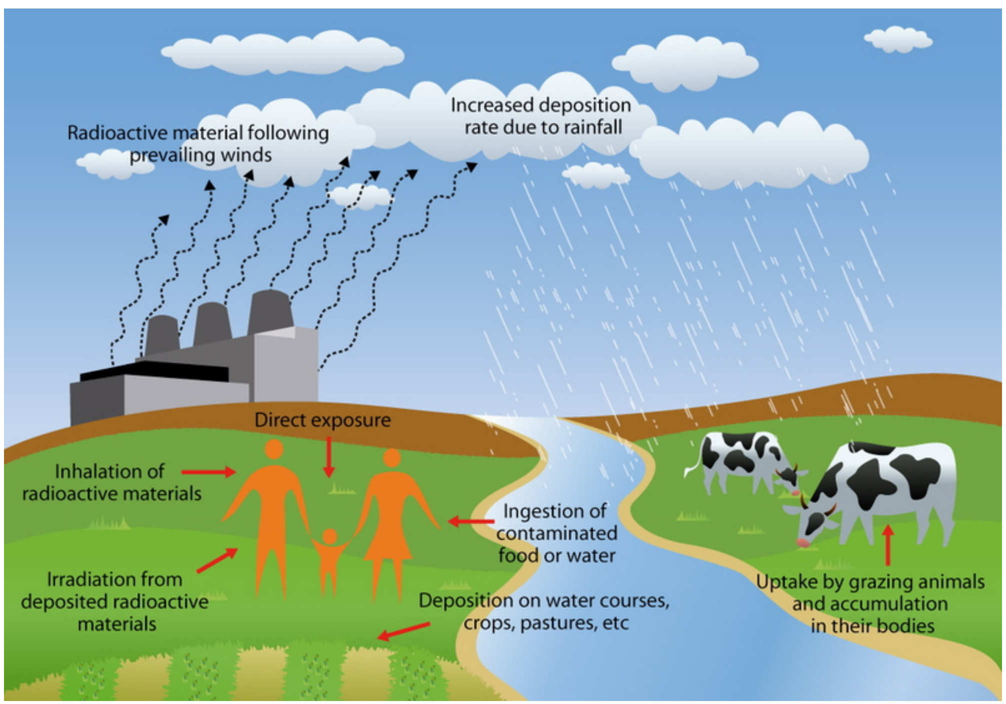
Consequence Analysis (MACCS) Code Development Plan for Non-LWRs

Jonathan Barr
Acting Chief, Accident Analysis Branch
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

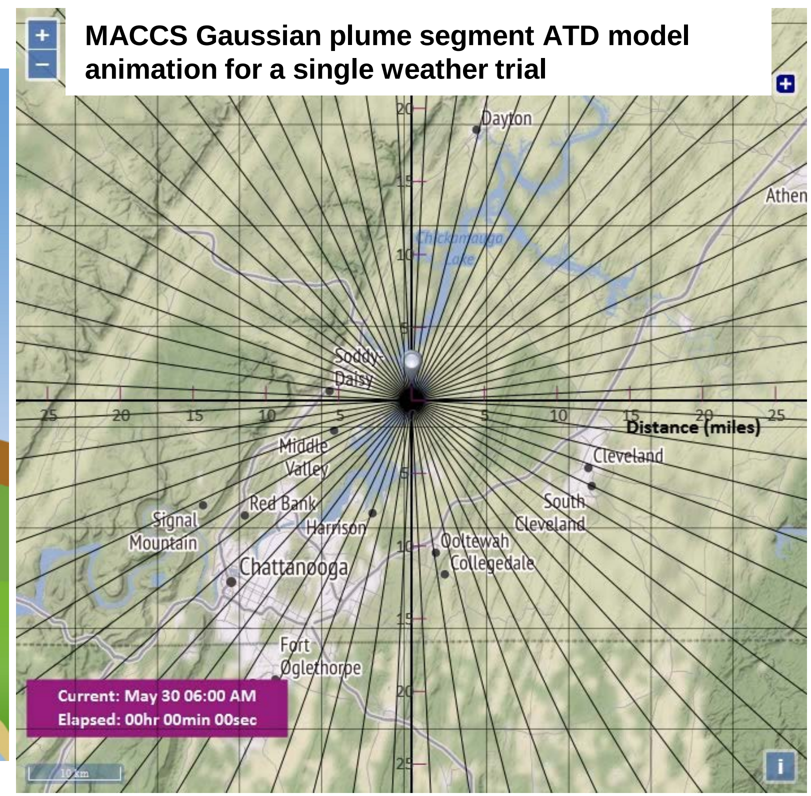
May 1, 2019

MACCS Overview

- MACCS is the only code used in U.S. for probabilistic offsite consequence analysis
- Highly flexible code that treats all technical elements of Level 3 PRA standard: radionuclide release, atmospheric transport, meteorology, protective actions, site data, dosimetry, health effects, economic factors, uncertainty



MACCS Gaussian plume segment ATD model animation for a single weather trial



MACCS Applications

- Regulatory cost-benefit analysis
- Environmental report analyses of Severe Accident Mitigation Alternatives (SAMA) and Design Alternatives (SAMDA)
- Level 3 PRA
- Research studies of accident consequences
- Support for emergency preparedness and planning

MACCS for Non-LWR Consequence Analysis

- MACCS is well-suited for consequence analysis for non-LWRs
  - Highly flexible code enabling use for different types of sources, accidents, and modeling applications
  - Several hundred user-controlled input parameters
  - Generally a technology-neutral code

- Alternative codes with some similarity exist but for different applications
  - **RASCAL** for incident response dose calculations to inform protective action recommendations
  - **RADTRAD** for design basis accident dose calculations for compliance with NRC siting criteria and control room habitability

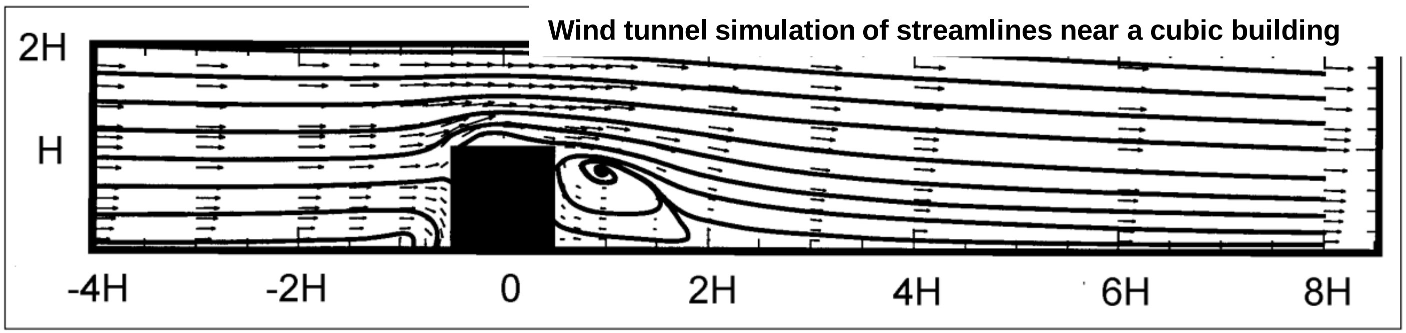
- MACCS compared to RASCAL and RADTRAD
  - Considers full range of protective actions
  - Calculates full set of consequence measures
  - More flexible for adapting to variations in severe accidents
  - Supports probabilistic application and impact of weather uncertainty

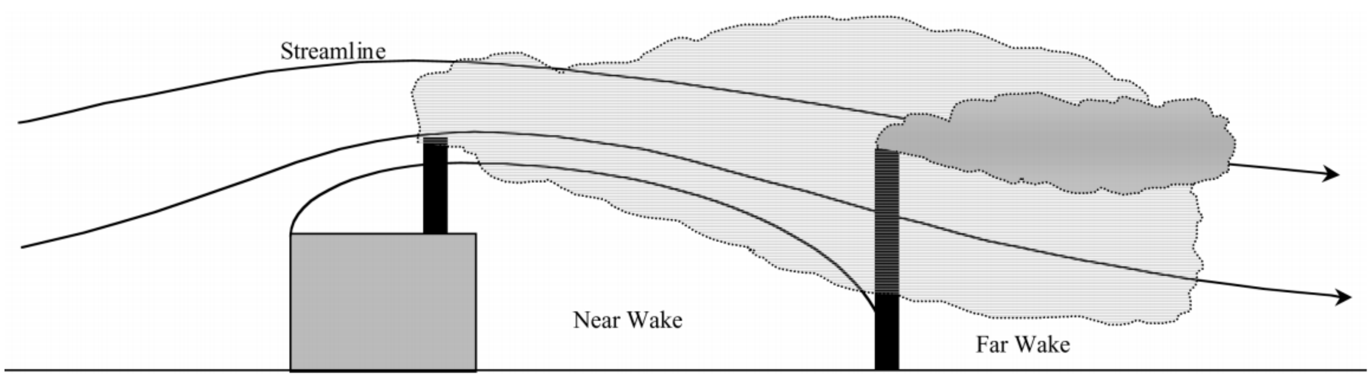
MACCS for Non-LWRs

- Code development plans for design-specific issues
  - Radionuclide screening
  - Radionuclide particle size distribution
  - Radionuclide chemical form
  - Radionuclide particle shape factor
  - Tritium
- Code development plans for site-related issues
  - Near-field atmospheric transport

Near-Field Atmospheric Transport

- MACCS currently has a simple model for building wake effects; user guide cautions against use closer than 500m
- Non-LWRs (and SMRs) desire smaller EPZ and site boundary than large LWRs; therefore desire better modeling of near-field phenomena



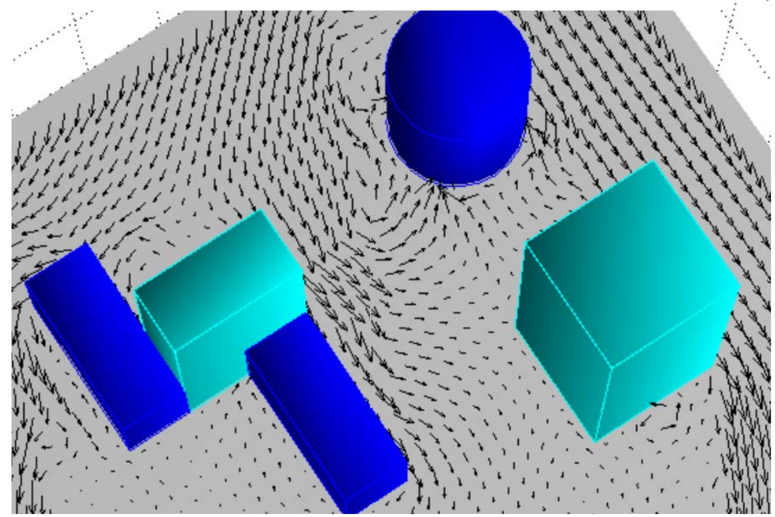


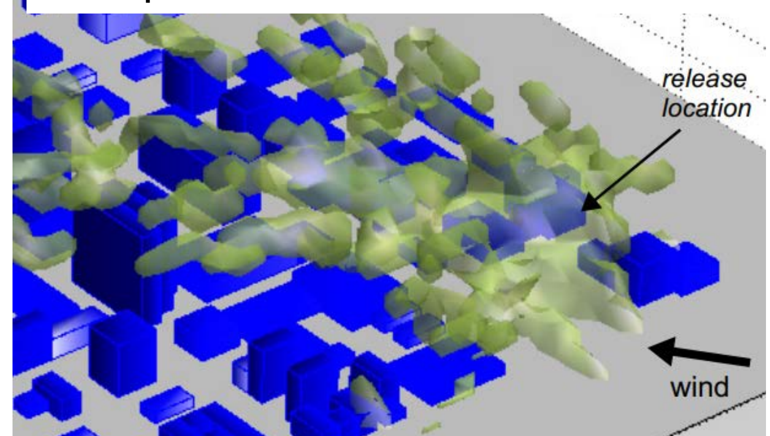
Lloyd L. Schulman , David G. Strimaitis & Joseph S. Scire (2000) Development and Evaluation of the PRIME Plume Rise and Building Downwash Model, Journal of the Air & Waste Management Association, 50:3, 378-390

Near-Field Atmospheric Transport

- Various options for addressing near-field ATD
  - Modifications to Gaussian plume segment ATD model
  - CFD modeling of 3-d wind field with Lagrangian particle tracking ATD model
  - Empirical models of 3-d wind fields with Lagrangian particle tracking ATD model

- Considerations for evaluating options
  - Extent of practical acceptance in the user community
  - Simplicity of use
  - Computational efficiency
  - Cost and time efficiency
  - Accuracy
  - Feasibility for probabilistic application


Example QUIC-URB simulation of wind vectors



Example QUIC-PLUME simulation of urban transport and dispersion

QUIC Factsheet, Los Alamos National Laboratory

Near-Field Atmospheric Transport

- Current status
  - Reviewed several existing ATD models that are relatively simple modifications to a Gaussian for building wake effects
  - Perform a near-field comparison of MACCS to alternatives to identify whether MACCS is conservative and over what distance range, conditions, and input parameters
    - Using existing MACCS with a point source, no buoyancy, and ground level release
  - Based on comparison, then integrate a Gaussian-based alternative into MACCS for better building wake effect treatment

BACKUP (MELCOR)

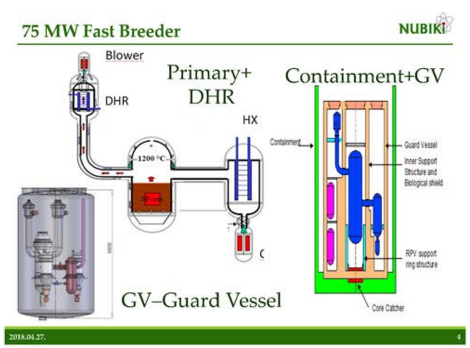
Example HTGR

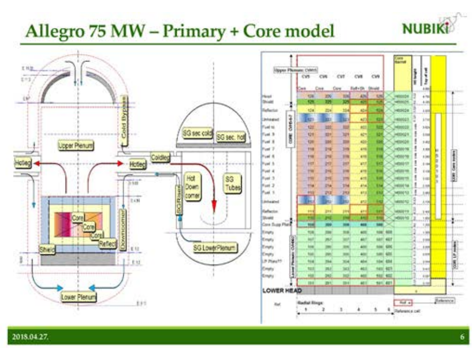
IAEA CRP-6 Benchmark – Codes (IAEA TECDOC-1674)

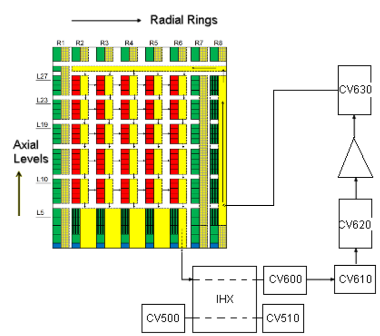
Case	1a	1b	2a	2b	3a	3b
US/INL	0.467	1.0	0.026	0.996	1.32E-4	0.208
US/GA	0.453	0.97	0.006	0.968	7.33E-3	1.00
US/SNL	**0.465**	**1.0**	**0.026**	**0.995**	**1.00E-4**	**0.208**
US/NRC	0.463	1.0	0.026	0.989	1.25E-4	0.207
France	0.472	1.0	0.028	0.995	6.59E-5	0.207
Korea	0.473	1.0	0.029	0.995	4.72E-4	0.210
Germany	0.456	1.0	0.026	0.991	1.15E-3	0.218

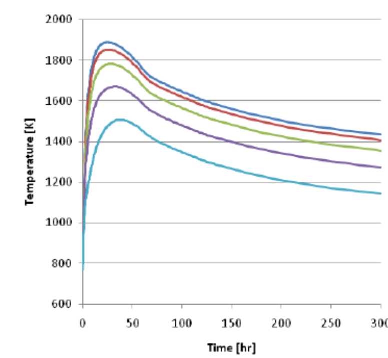
(1a): Bare kernel (1200 °C for 200 hours)
(1b): Bare kernel (1600 °C for 200 hours)
(2a): kernel+buffer+iPyC (1200 °C for 200 hours)
(2b): kernel+buffer+iPyC (1600 °C for 200 hours)
(3a): Intact (1600 °C for 200 hours)
(3b): Intact (1800 °C for 200 hours)

Plant Model and Accident Analysis



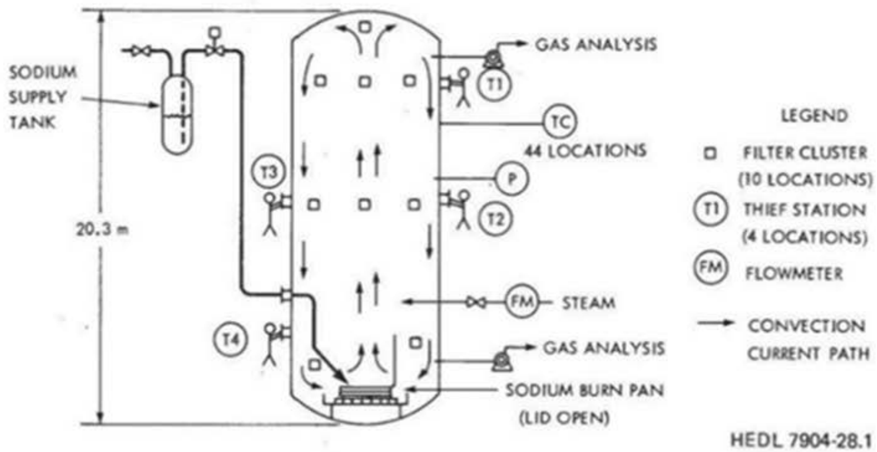


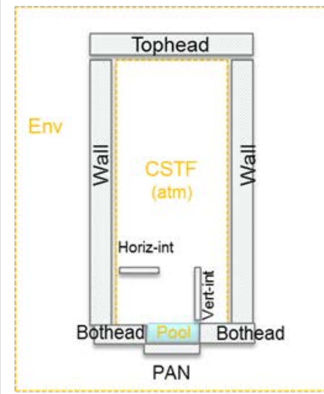




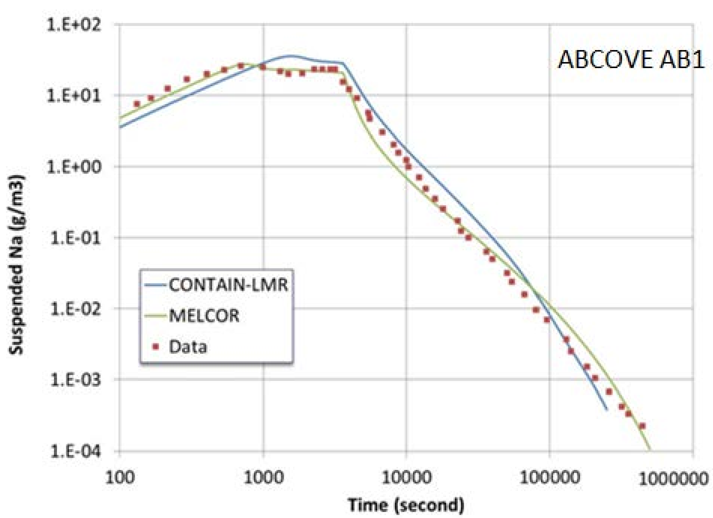
Example SFR

Containment System Test Facility (CSTF) & MELCOR Model

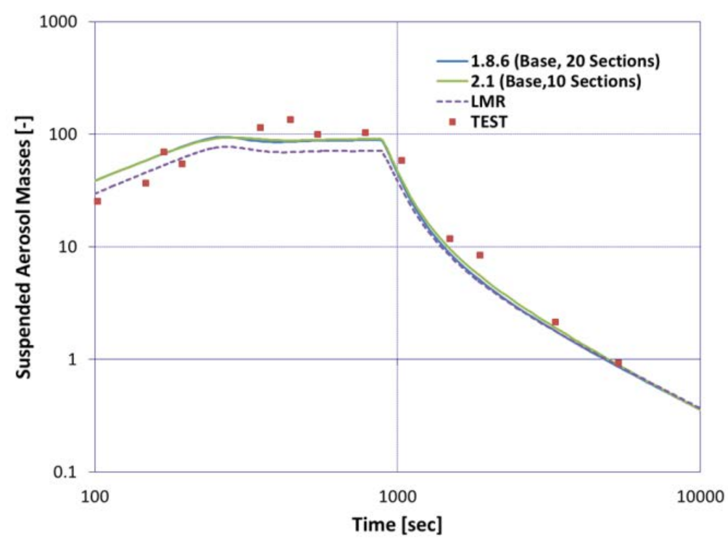




Sodium Pool Fire Test AB1



Sodium Spray Fire Test AB5



MELCOR Design Objectives & Success Criteria

Requirements for Integrated Severe Accident Analysis

- MELCOR predictions of phenomenological events are in qualitative agreement with the current understanding of the physics of such events based either on the results of certain well-defined/controlled experiments or on analytical results derived from first principles.
- Uncertainties in key parameters describing a phenomenon as calculated by MELCOR are in quantitative agreement with the uncertainties in experimentally measured or analytically derived values of these parameters.
- Where feasible, MELCOR phenomenological models are mechanistic in nature and capture the major physical processes. Alternatively, parametric models are used and uncertainties in the phenomena can be adequately represented through parametric variations and sensitivity analysis.
- Code user guidance is available to facilitate and standardize plant calculations of targeted applications in seeking consistent and reasonable key figure of merit predictions.
- All plant input models/applications and code assessments are well-documented, and non-proprietary documents are available to users.
- MELCOR is portable, robust, and relatively fast-running.
- The maintenance of the code follows software quality assurance standards for configuration control, testing, and documentation.
- To the extent possible, modeling will be simple rather than complex and technically consistent within the lumped parameter code framework.

Summary

- Staff is assuring that codes and methods will be ready to support licensing
- Staff is actively preparing our analytical codes and methodologies to ensure NRC's readiness to support licensing of non-LWR designs
  - Technology specific studies
  - Code development activities to support confirmatory analyses

Next Steps

- Seeking feedback
  - What are the most significant modeling and simulation "gaps" for non-LWRs and are they appropriately accounted for in the reports? Are important "gaps" omitted?
  - Does this Strategy represent the best course of action to develop sufficient expertise and readiness for conducting non-LWR reviews? If so, why? If not, what changes to the Strategy should be considered?
  - As an alternative, should the NRC consider using developer and applicant codes for confirmatory or sensitivity analyses? What are the pros and cons to this alternative approach?
  - What should the role of the NRC's codes be using a Licensing Modernization Plan approach?
- Additional technical meetings, as requested