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UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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714TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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OPEN SESSION

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WEDNESDAY, APRIL 3, 2024

The Advisory Committee met via hybrid In-person and Video-Teleconference, at 8:30 a.m. EDT, Walter Kirchner, Chairman, presiding.

COMMITTEE MEMBERS:

WALTER L. KIRCHNER, Chair

GREGORY H. HALNON, Vice Chair

DAVID A. PETTI, Member-at-Large

RONALD BALLINGER, Member

CHARLES H. BROWN, JR., Member

VICKI M. BIER, Member

VESNA B. DIMITRIJEVIC, Member*

JOSE MARCH-LEUBA, Member

ROBERT P. MARTIN, Member

THOMAS E. ROBERTS, Member

MATTHEW SUNSERI, Member

ACRS CONSULTANT:

DENNIS BLEY*

DESIGNATED FEDERAL OFFICIAL:

HOSSEIN NOURBAKHSH

ALSO PRESENT:

STEVE BAJOREK, RES

BRADLEY BEENY, Sandia National Lab*

LUIS BETANCOURT, RES

ANDREW BIELEN, RES*

SHAWN CAMPBELL, RES*

KEITH COMPTON, RES

JAMES CORSON, RES*

HOSSEIN ESMAILI, RES

LUCAS KYRIAZIDIS, RES

MIKE SNODDERLY, ACRS

JOHN TOMON, RES

CASEY WAGNER, Sandia National Lab*

KIM WEBBER, RES

* present via video-teleconference

AGENDA

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

8:30 a.m.

CHAIR KIRCHNER: Good morning. The meeting will now come to order. This is the first day of the 714th meeting of the Advisory Committee on Reactor Safeguards.

I'm Walt Kirchner, Chair of the ACRS. Other members in attendance are Ron Ballinger, Vicki Bier, we expect Charles Brown, Vesna Dimitrijevic, Greg Halnon, Jose March-Leuba, Bob Martin, Dave Petti, Thomas Roberts, and Matt Sunseri.

I will also note our consultant, Dennis Bley, is with us remotely, and I also note that we have a quorum. Today, the committee is meeting in person and virtually. The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act.

The ACRS section of the U.S. NRC public website provides information about the history of this committee and documents such as our charter, bylaws, Federal Register notices for meetings, letter reports, and transcripts of full and subcommittee meetings, including all slides presented at those meetings.

The committee provides its advice on safety matters to the commission through its publicly available letter reports. The Federal Register notice announcing this meeting was published on March, and I don't have the date. I'm sorry. This announcement provided a meeting agenda as well as instructions for interested parties to submit written documents or request opportunities to address the committee. The designated federal officer for today's meeting is Hossein Nourbakhsh.

The communications channel has been open to allow members of the public to monitor the open portions of the meeting. The ACRS is inviting members of the public to use the MS Teams link to view slides and other discussion materials during these open sessions. The MS Teams link information was placed in the agenda on the ACRS public website.

Periodically, the meeting will be open to accept comments from members of the public listening to our meetings. Written comments may be forwarded to Hossein Nourbakhsh, today's designated federal officer.

A transcript of the presentation portions of the meeting is being kept, and it is requested that speakers identify themselves and speak with sufficient clarity and volume so that they can be readily heard. Additionally, participants and members of the public should mute themselves when not speaking. And let me just amend my comments to say that the Federal Register notice with the agenda was published on March 14, 2024.

With that, today we are going to consider a number of topics, starting with we'll continue our review of the NRC research programs with a presentation on non-LWR code development. This afternoon, we will hear report outs from members on the NuScale SDAA, and tomorrow in our planning and procedures meeting, we will continue our preparation for our presentation to the commission scheduled for June.

So, with that, I'd like to turn to other members and see if you have any further opening remarks. Hearing None, then I will turn to Dave Petti and Bob Martin to introduce today's topic. Is it Bob or Dave? Bob, okay, Bob Martin.

MEMBER MARTIN: I'm Bob Martin, and on behalf of the Safety Research Subcommittee, the, you know, kind of cherry today. As Walt noted, we'll be talking about non-light water reactor computer code development. And this is just one meeting in a series of NRC research topic meetings that the ACRS will be hosting over the next year, culminating early 2025 as part of our triannual review of NRC research activities.

The focus of today's meeting is the NRC research report entitled NRC Non-Light Water Reactor Vision and Strategy, Volumes 1 through 5, covering the following topics related to non-light water reactor computer code development: plant systems analysis, fuel performance analysis, severe accident progression, consequence analysis, licensing and siting dose assessment, and nuclear fuel cycle analysis.

In addition, included in the material provided for this meeting is a supplement document entitled Status Update on Computer Code and Model Development for Non-LWRs.

It is my understanding that the NRC research near present has asked our committee to prepare a letter expressing our perspectives related to the completeness of the work and its future plans as it relates to NRC safety missions.

To this end, the committee will gather information, analyze relevant issues and facts, formulate proposed decisions and actions as appropriate, and we have scheduled time during our May full committee meeting to finalize the requested letter.

And now at this time, well, I guess you've already kind of opened it up for remarks from everyone else. I guess with no further, say, member remarks, we'll just proceed. And as I've noted, we'll hear on several subjects related to non-light water reactor code development.

The published agenda for today has us going to mid-afternoon with a 60-minute recess for lunch, and of course, appropriate breaks, you know, based, of course, on how the flow of the conversation. I'd like now to call on Kim Webber, Division Director of Systems Analysis in the Office of Research, to make introductory remarks and anything else.

MS. WEBBER: Yes, thank you for that nice introduction, and good morning to all of you. Thanks for taking the time to review our most recent report that documents the progress of our code development activities as it relates to supporting licensing for non-light water reactors.

There are two reports that we submitted to the committee for review. One is called Progress Towards Code Development in Support of the NRC's Regulatory Activities for Non-Light Water Reactors, and the other is called Verification and Validation of the Comprehensive Reactor Analysis Bundle, BlueCRAB Report.

My name is Kim Webber. I'm the Director of the Division of Systems Analysis in the NRC's Office of Nuclear Regulatory Research. We're really happy to be here today even though it's a rainy, cloudy day. I'm very happy to be here to talk to you about the significant progress that we've made over the last several years to develop our staff expertise and also the analytical capabilities to support licensing of non-light water reactors.

Our meeting is the third of a series of meetings being led by the Office of Reactors as part of the triannual review of the NRC's safety research program, and as with all ACRS reviews of the program, you know, we would appreciate feedback and the final letter that you mentioned, so that would be really helpful to us.

In my overview presentation -- can we go to the next slide, please -- I'll briefly introduce the five branches that are in the Division of Systems Analysis, provide a short history of the efforts that we've been undertaking, and summarize some of the major ACRS conclusions and recommendations, and then my staff and branch chiefs will make presentations describing the progress we've made over the last several years, and I'll wrap-up with some conclusions at the end of the meeting. Next slide, please?

So, with me today are the branch chiefs and staff who have contributed substantially to the successes that you'll hear about during the meeting. Before I get started, I wanted to note that we have five branches in my division as shown on this slide.

The technical breadth of the division includes fuel performance, reactor systems analysis, source term, accident progression, accident consequences, radiation protection, and health physics.

The names of my branch chiefs are noted here below the name of the branch, and also identified are the lead branches and branch chiefs for the various volumes that are included in that orange-colored row at the bottom.

I want to express my sincerest gratitude to all of my staff and their contractors for planning and doing the hard work to achieve the successes that we've attained to date. Next slide, please?

So, to facilitate the agency's readiness, the NRC's near-term implementation action plan -- you can skip through the other -- there you go. The near-term implementation action plan was completed in the summer of 2017 by NRR.

The IAP is the vehicle to execute the NRC's vision to safely achieve effective and efficient non-light water reactor mission readiness. The IAP includes six strategies, and strategy two on computer codes and knowledge to perform regulatory reviews is the focus of today's presentation. Next slide? You can skip again. There we go.

In March 2021, we completed a set of six reports, which you can see on the left side of the screen, and those included an introduction and five volumes that identify computer codes we plan to use for our independent safety analysis. They contain information about gaps, code development capabilities and data, verification and validation needs, along with specific code development tasks and methods.

Each of the volumes is focused on a different type of safety analysis capability, including reactor systems in volume one, fuel performance in volume two, severe accident progression source term and accident consequences in volume three, licensing and siting dose assessment in volume four, and front-end and back-end of the fuel cycle considerations in volume five.

I'd like to thank the ACRS for conducting in-depth and thorough reviews of our plans and progress over the last several years, which has significantly contributed to our success, I believe.

During today's meeting, you'll hear about our code development progress, which is documented in a single report that you can see on the right side of the screen. So, we'll no longer be updating the individual documents as we go forward. We'll update the document on the right side of the screen that you see occasionally. Let's go to the next slide? This is one of my favorite slides.

I thought I'd take a few moments to summarize our interactions with the ACRS since 2018 and to highlight key conclusions and recommendations as documented in several letters over the last few years.

In 2018, there were two ACRS meetings, one with DOE and the other primarily with industry, and that was focused on information about the DOE Office of Nuclear Energy-funded code development programs, which at that time included The Hub or the Consortium for Advanced Simulation of Light Water Reactors, or CASL, and the Nuclear Energy Advanced Modeling and Simulation or NEAMS programs.

Following those meetings, we participated in eight ACRS meetings to describe our code development plans and progress at various stages. In the back-up slides to this deck, there's a really nice synopsis of the meetings and links to associated documents.

I've paraphrased many of the ACRS conclusions as documented in the letters, which also dovetail nicely with, I think, our assessment of where we're at today. In general, the approach we've taken has been to update NRC codes like SCALE, MELCOR, and MACCS, and the licensing and siting dose assessment codes, plus leverage DOE codes to fill computational gaps in NRC's reactor systems analysis codes.

Since 2018 when we started in earnest to build out these tools, we actively followed the priorities of the non-light water reactor community and industry, DOE funding streams, and feedback from NRR to prioritize budgeted resources for code development activities.

A key aspect of our success has been to leverage NRC DOE memorandum of understanding to gain access to the deep technical expertise and other resources at the National Laboratories. We're extremely grateful for the opportunities to collaborate with our colleagues at DOE and the labs. We've been given access to the whole suite of the NEAMS codes, along with many training opportunities cost free. Additional collaborations with personnel in DOE's National Reactor Innovation Center have yielded the cost-free development of many reference plant models that complemented the NRC's existing libraries.

In general, we feel that we've got the capabilities to perform independent confirmatory analysis when requested. Having design-specific information and the time to update the codes will ensure that they produce reasonable results, which will support shorter schedules for non-light water reactor reviews.

Later in the presentation, you'll hear about a recent success we've had in using our codes to support the Hermes construction permit application technical review. That work was presented at an ACRS meeting and I think you'll be familiar with it when you hear about it.

For many of the codes and code suites you'll hear today, we embarked on a plan to use publicly available plant design information to build what we call reference plant models. They have been used to test and verify the codes, identify information and data gaps, and help train the staff, which has been critically important.

The shift, I'm sorry, the siting and licensing dose assessment area, we consolidated many of our codes into a new code called SIERRA, which you'll hear about later, and we also undertook an effort to assess analytical capabilities in our SCALE and MELCOR codes to support licensing the front and back-end of the fuel cycle.

For many of the codes, we've completed a significant amount of code validation, although there is still more work to do. For some non-light water reactor designs, there has been more experience and data, such as for sodium fast reactor technologies, although for other non-light water reactor designs, there is much less experience and much less data that's available to us.

We believe we could do a reasonable job using our codes to assess the margins relative to safety limits and key figures of merit, and also to characterize uncertainties. An equally important aspect of the work on our codes over the last few years has been to build staff expertise, enhancing knowledge related to the designs, operation, and accident sequences for non-light water reactors.

Many thanks go to Idaho, Argonne, and Sandia National Labs, as they've led many formal and informal sessions for the staff and public to train on the BlueCRAB suite of codes and also conduct public code demonstration workshops. Next slide?

Regarding ACRS recommendations, I note a few on this slide. Overall, I think we have a broad range of analytical capability to support NRR's request for less detailed safety studies, such as to demonstrate how a new reactor design may operator, or requests for more detailed confirmatory analysis for situations where there are small margins or large uncertainties. As an example, I referenced the Hermes construction permit application in which MELCOR was used to help understand the progression of certain accident sequences.

As I mentioned previously, we've used reference plant models with our codes to perform pilot studies and perform demonstration calculations, such as was done with SCALE, MELCOR, and MACCS. We are also performing pilot studies using reference plant models with the BlueCRAB suite of codes.

Regarding the last recommendation, which was made several years ago, identification of the level of effort of the licensing reviews is really not the role of the Office of Research. NRR has successfully demonstrated their use of core teams and newly-developed guidance to appropriately size the level of effort for their reviews. They also have made many presentations to ACRS on a wide range of topics, including the Kairos Hermes construction permit review, Part 53, ARCAP, and microreactors.

And so, now let me introduce Steve Bajorek. Steve is no stranger to the ACRS and has presented many times. He's our senior level advisor for thermal hydraulics and he'll lead the next part of the presentation.

MEMBER MARTIN: Before you make that transition --

MS. WEBBER: Sure.

MEMBER MARTIN: Historical context, and the executive summary mentions the 2016 commission has a vision statement on --

MS. WEBBER: Okay.

MEMBER MARTIN: So, it's eight years --

MS. WEBBER: Yeah.

MEMBER MARTIN: -- since that has happened.

MS. WEBBER: Okay.

MEMBER MARTIN: I guess I'd be curious as, you know, as the day goes on, any insights on whether, you know, what extent maybe that vision statement needs to be updated given all the water under the bridge --

MS. WEBBER: Yeah.

MEMBER MARTIN: -- at this point. Certainly, you've done a lot, you've learned a lot, and that synthesis of that experience should translate into the next vision statement.

MS. WEBBER: Correct.

MEMBER MARTIN: But keep that in mind, and it's kind of a question you could ask anytime, but obviously at the end, maybe we'll be a little worn out, so --

MS. WEBBER: Yeah, the one thing I'll say right now is that in large part, we're ready now. So, over the last eight years, the focus has been on getting ready, but now we have the capabilities, we have the experience, so we're ready now.

And I think, you know, our counterparts in NRR feel the same way. You know, with the regulatory strategies, they feel like we're ready now. So, there's probably an appropriate need to update that, although that document was a point in time.

MEMBER MARTIN: Right.

MS. WEBBER: And it's not -- you know, it took a lot of work and resources to update that document or prepare that document, and with all of the flurry of activity going on, I'm not sure that that's a very high priority given everything else that NRR needs to accomplish. So, anyway, I just put that out there.

MEMBER MARTIN: I guess the side thought I had to that, the timing of 4068. It was a little bit before this middle switch towards more risk-informed, performance-based approach to things, maybe a little more emphasis on source term and dose consequences than we've had historically, and so it's a little bit of a different flavor of an emphasis on how to use these codes. That's really what was in the back of my mind when --

MS. WEBBER: Yeah.

MEMBER MARTIN: -- when I mentioned that, but I do appreciate everything that I've seen and the work you've done that you've provided us. You know, it seems very thorough and I can see where you've come with this comment that we are ready now.

MS. WEBBER: Yeah, one last thing I would like to mention, so the NRC is in what I would consider a very unique position to have to perform safety analysis for all the different kinds of non-light water reactors that come in for review, whereas an individual vendor or applicant, they're focusing their codes and their capabilities on one particular design.

So, we have that added complexity of having to be ready to look at that wide range of technologies that we'll likely receive in applications over the next coming years, so thanks for the comments. Steve?

MR. BAJOREK: Thank you, Kim, and first let me share my screen. And the slideshow from the beginning, and I don't have to ask because I can see the slides right up there.

Anyway, well, good morning, everyone, and it's a real pleasure to be back and to be able to brief everyone on the progress we've made, which we call the volume one system analysis codes which we've been calling BlueCRAB.

There are four things that I'd really like to accomplish this morning. First, I want to go back through a little bit of background information on how we got to where we're at today. Why did we come up with BlueCRAB and why do we feel that we need to go in this direction to look at the non-LWRs? In particular, how do we see its use in the review and whether it's going to be those intended applications?

I know we've had -- there are several new members that may have not been for those original meetings, and it's been a good five years since we've talked about this, so I want to spend a few minutes on that.

When we were starting off five, six years ago, one of the questions was validation, how much assessment's been done, and the answer at that point was well, there's some out there. We didn't know where it was at because in one case, rather than being like in a vendor as Kim pointed out, it's spread around. The labs are doing different codes. Some of them are being assessed by the NEUP program, some by the NRC. It's not all in one place.

So, what we started about a year ago is let's put together a V&V report that would at least put a wrapper around what has been done, and use that as a way and a means to identify what else needs to be done before we can move on. So, I'm going to talk a little bit about the V&V report, what's in there, what the status is, and I think it's interesting as you go from technology to technology, you get a better feel for what's more mature than some of the others.

And then I'm going to go into some of the reference plant models that we've developed with BlueCRAB. There are six of them that we're actively working on. I can't go through all of those in a whole lot of detail.

I can take any one of these models and we could spend at least two or three hours on it to look at what we've done, how we model it, what the issues are with each one of those, but I want to give a status on where we're at, what we've done, and indicate where we need to go. So, I'm going to spend some time on that, and then we'll wrap-up with a summary and some next steps.

But before going on, I really got to put an acknowledgment out there to colleagues at Argonne National Lab and Idaho National Lab. This has been a coordinated effort over the past several years.

We really have to compliment Rui Hu and his coworkers at Argonne, principally in the area of the thermal fluids' development that we're doing, Javier Ortensi and his coworkers at Idaho with his work on the Griffin Code and the neutronics. Putting it together has been -- we've had to have the labs work together and we've had to work together to come to an understanding on how this all should come together.

As Kim mentioned, one of the biggest benefits working with the labs has been the expertise that we've been able to gain from them. When we develop a reference model, we bring it in-house, but we have a hands-on workshop, multiple workshops, where either remotely or people come here, we get the people who are working on it at the NRC to take the codes, run them, okay, understand the model, adopt those, ask questions on what's going on within the model and how they're getting some of the results that we're getting.

And over the last few years, I think I'm very pleased to see that we're now starting to get a sizeable number of staff members here at the NRC that understand the codes. They understand how they go together. They can independently take those things and make some changes. We still need a lot of help on that.

And more importantly, when you take those tools and you use them, you start to really understand the technology. As we've gone from model to model, that has been vital because as we pointed out five, six years ago, we were a very water-centric organization.

We understand light water reactors for the most part. There's still some questions on that, but except for the work that had been done in gas cooled reactors with NGNP, we were a little bit behind on there. So, I really have to point out the contributions that everyone has helped us with.

Now, volume one is about systems analysis. We want to be able to analyze the entire system, the various conditions that we might encounter. In volume one, if we go back and look at it, we were at the state of defining what codes we felt we needed to use with the non-LWRs.

We started that off using our EMDAP process, evaluation model development and assessment, by first going through the available PIRTs at the site, phenomena identification and ranking tables. What are those phenomena out there that are new and different, things that would give us challenges, both with the NEAMS codes that we're using and with the NRC codes? Should we have gone and tried to develop them along those lines? And also use those PIRTs to help identify where there are shortcomings either in the experimental database or our knowledge base.

The intended applications for the BlueCRAB codes are first obtain steady state conditions with a fair amount of detail, power, both radial and axial, temperature distributions, velocity, flow distributions within the core of the vessel, the entire system, primary, secondary, tertiary systems if we need to go in that direction.

That's your starting point, but then we would move onto accident analysis for scenarios that don't result in core disruption, okay, but for these types of designs with the margins that are being claimed and proposed, this would be unprotected loss of flow, unprotected loss of heat sink.

LOCAs for the most part have been designed out of the system, but we would look at those, reactivity insertions, heat pipe failures, this whole gamut of things that really help us understand how the machine works if it operates based on the applicant's claims, and this is where the staff education really comes into play because we want to make sure that we understand the system. If there is an offset, a problem, a scenario, we understand what goes on and how that system should mitigate it.

MEMBER MARTIN: Question.

MR. BAJOREK: Sure.

MEMBER MARTIN: So, Kim noted in her

overview the biggest challenge, of course, in preparing for, you know, non-light water reactor applications is the fact that you don't know exactly what you're going to get.

Now, the nature of PIRT and EMDAP is that when you're doing the PIRT, you've kind of already paired up a particular plant and even a particular event, that you've already kind of embedded the scenarios. I mean, don't you expect some gaps inherently? And in the eight years, I guess, you've been looking at this, I mean, have you identified those?

MR. BAJOREK: Oh, absolutely. I think EMDAP takes you through the whole process. The PIRT kicks things off. It doesn't always solve things for you, okay? It helps you get started. Assessment is part of that, but as we go through this, we have to make sure that we have consistency in the assessment and how it's being used in the plant model.

We think in the long run, we will probably need to look at some uncertainty methodologies because we have a lot of these phenomena that we haven't investigated to the degree that we have in light water reactors. We have some things out there like viscosity in a molten salt reactor that you don't know within plus or minus 20 percent. How that does that effect your analysis?

So, even though we have done the PIRTs, we've done the assessment and things like this, we need to make sure that we go back, and as we look at a particular application, it all applies. Scaling has to be brought into there. We have to look at the uncertainties.

And at this point, we only can deal with the known unknowns, okay? Things like solidification, we didn't have that in the codes from the start. It's in there now, okay? So, we can deal with those things as we see it.

But if an applicant comes in with a particular type of reactor cavity cooling system that we haven't encountered before, a DRACS system that we haven't really looked at, or a type of geometry that is out there -- a lot of these plants now are going away from a loop type design, pump things out of the system through a heat exchanger, back to a pool type design where it's all within the vessel.

That gets rid of LOCA, but it puts a greater burden on you to understand natural circulation and a complex geometry. We haven't faced that yet, and when we see these new designs, those are some of the questions that we're going to have to look at.

MEMBER MARTIN: So, you mentioned uncertainty resonates because, of course, the points you made there are inherent in this work. There are a lot of uncertainties. In the previous generation of computer codes, obviously just like the current generation, all of the emphasis was on development, just trying to get the physics down, and then the hooks to be able to actually do, say, a best or plus uncertainty, we'll put in afterwards.

Is that the same situation we have here today, that maybe the codes that you all are working with do not have the capability to really incorporate, say, a plus or minus 20 percent on a core heat transfer or something like that? I mean, where --

MR. BAJOREK: It's sort of a mixed bag. There has been some work, but we're also going to be dealing with a coupled multi-physics environment.

MEMBER MARTIN: It makes it even harder.

MR. BAJOREK: And in some of these, it's the reactor dynamics, the neutronics which is going to maybe dominate the uncertainty along with things like how well we can model the thermal fluid environment or even the tensor mechanics in a fast reactor. So, there's still work that's going to need to be done in uncertainty analysis that I think is going to be more complex than what we had to deal with --

MEMBER MARTIN: Right.

MR. BAJOREK: -- on light water reactors.

MEMBER MARTIN: Well, I think, to throw in my personal opinion, this is why the, you know, lower fidelity, I hate to call system codes lower fidelity because they were the high fidelity ones, but the value, of course, to be able to be agile and to incorporate uncertainties, of course, has been tremendous for the industry, which, of course, we have the best estimate methods for light water reactors. They give you insights that you just can't get.

The tools that, you know, a lot of the tools that DOE has, including the system codes, which are higher fidelity, I would say, than, say, the traditional on-a-volume approach that we've had, there's significant complexity making it difficult to incorporate uncertainties.

So, you need tiers of capability and, you know, I'm going to be sensitive to maintenance of old as well as, of course, bringing the new up to speed. I mean, that's always the challenge, so.

MEMBER MARCH-LEUBA: Bob, Dennis raised

his hand.

MEMBER MARTIN: Yeah, oh, okay, Dennis?

MR. BLEY: Yeah, I took it down, but I was just trying to reconcile this discussion and Kim's statement that we're now ready. We don't really need a plan for going forward. I expect she was intending that to be for the applications you expect maybe in this year.

But I think from what I've understood, you guys are really feeling competent with the codes as they exist now, but some of the things like the complex geometry for natural circulation, some of the places where we were very sparse on data and probably still are remain, and I don't know if you want to talk to that, Steven, or if that will come up later for somebody else.

MR. BAJOREK: I think we'll get to it later. I think what we'll say --

MR. BLEY: That's good.

MR. BAJOREK: I'm sorry, go ahead? Okay.

MR. BLEY: Go ahead. Yeah --

MR. BAJOREK: Okay.

(Simultaneous speaking.)

MR. BAJOREK: We think the state of the codes are good at this point, that we can do a, I'll call it a singular analysis. If you asked me to do an unprotected loss of heat sink for a system, we can go and do that, and we probably have enough assessment to feel confident that we're on the right path, but there's more work we can do because some of the uncertainties come in a couple of different ways.

One, we're dealing with a higher fidelity system of codes and they have meshing capability that we haven't had to really deal with a whole lot for the systems codes. That's a question mark. There's the scaling of that assessment data to the new design that's out there, and the uncertainties in the models and correlations, okay, we haven't really used that to the extent we can at this point.

There's nothing in our regulations that say you have to go that way, okay, so we think we can do it without it. My recommendation is that we absolutely incorporate that because I want to know how much that eats into the margin.

And secondly, with these uncertainty methodologies, you can point to things that dominate your uncertainty, and hopefully that's where we focus our reviews in the future and not just open it up to everything that's interested to these plants.

MEMBER MARCH-LEUBA: At this point -- this is Jose -- I'd like to bring back the discussion two or three years ago, the last time we talked, and if you remember our recommendation. In my mind, if these advanced reactors don't bring oodles and oodles of safety margin, you don't have no business bringing them in.

MR. BAJOREK: Yeah.

MEMBER MARCH-LEUBA: So, in a sense -- I mean, we have some reactors that say we don't even need control rods. You just let the reactor heat up and it will shut down itself. So, I don't think the problem is determining uncertainty or validity. And I'm with you.

I mean, we've worked together, Steve, and as a model, low fidelity and everything, but do we need it? Do we need the complexity? If you can get ahead with a back of the envelope calculation and worst case scenario for heat depth, why spend ten man years on a calculation?

MR. BAJOREK: Yeah, I mean, our approach with the volume one is to try to keep it simple, but not simplistic. Do we --

MEMBER MARCH-LEUBA: Yeah, good.

MR. BAJOREK: Yeah, but we're also exploring some other questions. We'll see it later on. Yes, we think we can probably do with a one-dimensional model of the core. That's probably good enough, but you can't answer that question unless you try it three-dimensional and see if it makes a big difference, so we're trying to --

We'd like to sort some of those problems out now and look at the complexity in the hopes that we can throw it away eventually. So, it's -- you know, we're not there yet, but hopefully when we get into production mode, we wind up with models and capabilities that are relatively simple and you can do lots of calculations quickly, okay, without the burden of the expensive overhead that you can build in.

MEMBER MARCH-LEUBA: I'm just being an old wise guy, although I don't look or I don't act like it.

MR. BAJOREK: Keep doing it.

(Laughter.)

MEMBER MARCH-LEUBA: There's a danger of getting lost in the roots of the problem, but where you need to go is see the line above the trees and see the forest.

CHAIR KIRCHNER: One way to look at it is what your role in support of the agency is, which is confirmatory work. The kind of thing where we are in the LWR business with best estimate, and uncertainty is all about chasing margin and extracting a little higher performance out of the existing systems and still have an adequate assurance, whereas here, we're dealing in many cases with first-of-a-kind, and it seems to me that, just as you said it, Steve, you can bound many ways the general physics of a problem.

If there are questions, it's more on the, the onus is on the applicant to demonstrate the safe performance of the machine, not necessarily the staff, and so that allows, as Jose was saying, you know, to step back and take the kind of approach that you're suggesting --

MR. BAJOREK: Yeah, we're --

CHAIR KIRCHNER: You don't have to design the machine for the applicant.

MR. BAJOREK: No, but the word flexibility came in and I just want to put this slide on here.

MEMBER MARTIN: I just want to throw in.

MR. BAJOREK: Sure.

MEMBER MARTIN: The applicants, to Jose's point, will more than likely have simpler models and incorporate a lot of uncertainties. If you don't have a tool they can incorporate uncertainties in the same kind of way, you kind of get an apples and oranges type of comparison, which, you know, more than likely, your kind of more best estimate should actually show more margin, but you're not really confirming the uncertainties if you don't have the capability.

Now, I'm going to say that I think 90 percent of the uncertainties probably could be handled somehow relatively easy. There's going to be just a few very important ones that you'll want to be able to pull out, like the heat transfer ones that are a bit more challenging that really probably need emphasis and explicit effort, you know, its own project to cover.

MR. BAJOREK: I agree. I'm anxious that we actually get into the review and do that work, because I think as we see in some of the reference models, there does appear to be a lot of margin in these technologies.

If we get something that we think is fairly close to where the applicant's going, it looks like, yeah, there is going to be sufficient margin, then that allows you a lot of benefits in your analysis. I can use some conservatism to bound things that I don't understand.

The only caveat on that is that if I'm a utility and if I see there's lots and lots of margin, I'm going to eventually try to find a way of using that margin. So, we have to look at the future on what happens when that margin does disappear, but we're well off on that.

MEMBER MARCH-LEUBA: The utilities have been having, knowing that two, three, five percent margin, because it's a lot of money for them. These reactors have two, three, 500 percent margin. They will not have the electrical generation capability to -- I'm changing the subject a little bit and I'm being nice to you.

In my mind, when the staff, NRC, reviews these advanced light water reactors, not light water, advanced reactors, the problem is not going to be the uncertainty of the calculation of your code. The problem is going to be the unknown unknowns. What have you not thought of?

And the only way the staff can do at least an attempt to do the review is be very familiar with the design, and you become very familiar with the design by having all of these codes, running them, and see what can possibly go wrong.

So, even though -- I mean, we've always said yeah, this reactor has the types of fuel, there is no way you can break it. Even if you take a hammer, it cannot break it. So, it has a lot of --

(Laughter.)

MEMBER MARCH-LEUBA: Wow, okay.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: But what else can go wrong? Therefore, we need to have the confidence that we understand the system, that you understand the system.

MR. BAJOREK: Right, its flexibility at being able to do lots of calculations, okay, and using that to identify here's a problem or there's a problem, and then if you need the detail, you go after that.

MEMBER MARCH-LEUBA: In those reviews, think outside of the box. It's important.

MR. BAJOREK: The reason I put this up here is just so that hey, we do need a lot of flexibility in our ability to model lots of different systems. Our mission right now is to be ready for any of them and all of them within the next two years, and that was also one of the driving points. We don't have that time to take our old NRC codes and develop them to do all of this.

We needed to jumpstart this by adopting some of the NEAMS codes, but this kind of gives you a gamut of what we're faced with, which is an interesting and kind of fun challenge in a way because you have lots of different designs, lots of variations in those designs, and some of those, there are going to be those unknown unknowns because there's only a handful of those that we have enough public or private information to really see where they're going, and that is going to be one of the things for the future.

VICE CHAIR HALNON: So, Steve, back on that graph, I mean, these are all first-of-a-kind reactors, no operating experience. What role do you see or do you even see a role, a major role in these code developments for Nth-of-a-kind licensing? I mean, are we going to get to a point where we can just plug in the site-specific parameters and say it's good to go? I mean, that's obviously an extreme, but --

MR. BAJOREK: It depends on whether that Nth-of-a-kind is really like the first one or whether there are deviations from that. I would see that the use -- maybe I'm going back to my light water reactor days. It's like okay, we have it operating. Can we upgrade it? Okay, how can we --

VICE CHAIR HALNON: Right.

MR. BAJOREK: -- use that margin to improve the economics either by a power-up rating, or at least in the light water reactor world, looking at higher peaking factors, higher F delta H. I don't know what the equivalent is for non-LWRs.

VICE CHAIR HALNON: But what I anticipate is a future argument of what Nth-of-a-kind means --

MR. BAJOREK: Yeah.

VICE CHAIR HALNON: -- and what is the boundaries and definition of it? Because all of those unknown unknowns that you're designing extra margin for, you're going to reduce those as you get operating experience, and so the second-of-a-kind, third-of-a-kind, fourth-of-a-kind, when is Nth-of-a-kind? And it's at some point.

So, my sense is that the codes will become very, very important down the road, especially as the staff tries to figure out what's the most efficient way of licensing the Nth-of-a-kind? Now, somewhere in there, you get away from this is different to exactly or enough the same. So, anyway, just a thought as you go through this because a lot of work here. Maybe at some point, we'll get to plug and chug and --

MR. BAJOREK: It may be. You know, this also, I think if you go back and you look at like the auto industry about 1900, there were like 200 different makers, and then after a few years, it went down to a handful.

VICE CHAIR HALNON: Right.

MR. BAJOREK: You sort of suspect that might happen here, and once that consolidates to a particular design or design type, that's when we can maybe put -- maybe that's when we put more detail and more emphasis on getting higher accuracy because you see that one going forward.

CHAIR KIRCHNER: Well, Steve, I think, just one person's opinion, you know, each and every one of these proponents will come in with their first design, and you already hinted and what typically happens. They then look to take that margin and extract more power because there's an economic overlay to all of this, obviously. And the way I see it, I'm surprised -- let me -- we shouldn't lead the presenters, but I'll lead you back --

(Laughter.)

CHAIR KIRCHNER: -- at the RIC, because I think when you put the left-hand column aside where we have fairly mature codes and so on, and we're dealing with two-phase flow, as you were setting up your presentation, you're saying well, we're looking at, you know, design basis, no core disruption.

One of the things you said, I think it was at the RIC, is that these are all single-phase flow, and that indeed makes the computational modeling, at least from the thermal hydraulics part, simpler, more straightforward. You don't have the complications of two-phase flow.

But I suspect what will happen in answer to Greg's leading question is -- let's just assume that each of these are successful in getting a prototype first-of-a-kind plant out there. They will then come back to you and say okay, we've got this now, but we want to extract 50 percent more power or whatever, and then the capability of the codes that you need, we can pick on one, like LWRs.

They are going to be naturally limited in size if they follow the basic design approach that's used now where leakage is an important characteristic in shutting down reactivity insertion kind of events and such, and so they will be pushing against -- this comes back to uncertainty now. Then more sophisticated analyses that take into account things like uncertainties become more important because they'll push up on the envelope of their passive or inherent safety characteristics as they start trying to extract more power out of the machine.

MR. BAJOREK: I agree.

MEMBER PETTI: But you can also --

(Simultaneous speaking.)

CHAIR KIRCHNER: Their codes will be, not more robust, but more mature, more refined.

MEMBER PETTI: But also I think beyond just say power uprates, some of these concepts have advanced fuels down the line, advanced materials, and so it might not be they're going to change the total power, but they want to have a higher burnup like we see with the water reactor, or even higher linear heat rates, and step that way before they change sort of the fuel. So, there's multiple dimensions here in terms of how they're going to evolve their technologies, and that makes it really --

MR. BAJOREK: That's a good point because you've heard some applicants come and say well, we're going to run the reactor, run this for five years, and then we're going to truck it away. Well, if there's a lot of economic value at the end of five years, they're going to come and say how about year six, yeah seven?

CHAIR KIRCHNER: Yeah, yeah, 12 years, yeah.

MR. BAJOREK: So, it's -- you know, I wish we had a crystal ball and could see some of that, but, yeah, we're going to see a lot of issues that grow with time as we do those Nth-of-a-kind, which I hope we see.

MEMBER MARTIN: Code development in this area has been going on, not necessarily in light water, since before I was born, and the sustainability of these efforts, I think, is self-evident. All of the things that you're saying, you know, that we draw from light water reactor experience, I think we'll see it with non-light water. We're building these things with a lot of margin, almost deterministically. Eventually, we'll eat into those margins just like we have with light water. They'll be around for a while.

MR. BAJOREK: Okay, I want to move ahead, so I'm going to go through a couple of these slides pretty quickly here because I want to get to some other work.

When we were going through volume one, we identified a lot of phenomena that were, I wouldn't say they're really new. The phenomena has been around a long time, but they played a lesser role in the light water reactor world, but they were going to be very important for non-LWRs.

And this went everywhere from stratification, to striping that we saw on gas reactors, chasing the placement of neutron precursors in the fuel salts, solidification, which we almost never worry about in the light water reactors, to, I phrase it as 3D conduction and radiation, but it's really heat transfer through a complex structure to the environment. I lose my heat sink. I lose my flow. I can get rid of that energy just from the grounding.

So, those are things that, you know, our codes just weren't really equipped to do because you throw that away because we're conservative to ignore some of those, and that's how we wound up with the comprehensive reactor analysis bundle, BlueCRAB, blue for federal.

It's built around the MOOSE framework. MOOSE handles data transfers, numerical solutions. It handles some engineering, physics, tensor mechanics, and conduction, but it primarily handles the data transfers, the types of things that as code developers we don't want to spend a whole lot of time on because we want to deal with the physics.

And that physics is embodied in, we use SAM for the loop thermal hydraulics. We can use PRONGHORN is we need more detail. We haven't explored that yet because we're trying to stay more simple at first, but we're going to test that out, Griffin for the reactor dynamics.

And because we saw some systems that were going to use Rankine cycles, water-cooled RCCS, and we're real comfortable with our staff using TRACE for those types of designs, we said well, let's make TRACE part of the NEAMS environment by making what they call MOOSE wrapped. So, we can transfer information from these codes to TRACE so that as we have to deal with these other types of systems, we have a staff that already understands TRACE and we can model it that way.

We have two fuel performance codes which are part of the mix, BISON, which is part of the MOOSE framework. We have coupled of FAST code in on that. Now, what I'm going to show in the reference plants, we don't really use the codes for fuel performance. We use BISON for thermal mechanical expansion, but not fuel performance and how fission gas release occurs and thermal conductivity degradation. So, that's not really in our non-LWR work right now.

However, I do want to point out that hey, having FAST and BISON coupled through MOOSE to TRACE does give us some nice flexibilities as we're looking at situations in light water reactors where maybe we do want the detail that goes on in one of those codes and don't have to rely on relatively simplistic models within TRACE.

We also have incorporated Sockeye into the mix when we're looking at heat pipe performance. Five years ago, we said no, we're keeping Sockeye over on the side as a side thing, but it's come along enough over the last year that we have confidence that we can use it, although we have simplistic models built into SAM to do the same thing for the heat pipes.

Nek5000 CFD code, or NekRS, the more modern version, it's there. We're trying to avoid using it. We want to stay simple, but if we got to go to that, we'll do that.

You see SERPENT on there right now for doing cross sections. Our goal here over the next couple of years is to phase it out, incorporate SCALE Shift by cross sections into Griffin, or make use of the work that they're doing right now to build an MC-2 built into Griffin. So, SERPENT has been a convenient thing in the workflow process, but that's eventually going to go away.

MEMBER MARCH-LEUBA: And now for something completely different.

(Laughter.)

MEMBER MARCH-LEUBA: You're probably old enough like me to remember when vendors could only run their version of the code in a VACS system, or PDP, or on a deck.

MR. BAJOREK: Yeah.

MEMBER MARCH-LEUBA: I assume --

MR. BAJOREK: Cray.

MEMBER MARCH-LEUBA: I assume we have moved all of these codes to a cold environment.

MR. BAJOREK: Cold, yeah. I mean, that's a good point because one of the other questions, how are you going to run this stuff? You know, if you look at somebody running Nek5000 and they have a billion cells and thousands of processors, are you going to have to do that? The answer is no.

We typically run these references models on a MacBook Pro, a dozen processors. You know, I can go down to the mall here and buy one and start running things that night, okay? You know, it chugs along and it does fine. We can run it on our own RES GOV cloud where we can compile the codes on there, we can run those, and we've got hundreds of CPUs at our disposal.

Now, right now, a lot of our users like to go and use the HPC on demand system on INL, and they've been letting us use it, so we can go through that portal. It has a very nice interface and we run it on there. So, if we need the thousands of cores, we can use that.

We can also get that from our own Gov cloud, which is going to be important because we want to make sure proprietary information stays here. We don't want to ship that to the labs or anything. So, we're run it that way, and we can also run most of these problems right now on MacBooks. We haven't encountered anywhere it has been unfeasible to do that.

MEMBER MARCH-LEUBA: Yeah, where I was trying to go is obsolescence, specifically when you have these complicated, dense communications in different codes. When you have a single code and you compile it, and you have a single memory block, you're going to surely need it, but when you have things this complicated, are you designing the systems so that five years from now, it will still run?

MR. BAJOREK: I hope so.

MEMBER MARCH-LEUBA: Let's think about it.

MR. BAJOREK: The MOOSE codes have at least been parallelized to the extent possible. And they seem to be very portable.

TRACE, we could try to do that, to get there. But yeah, you're right, we have some of these serial codes, a one-on-one processor, and if you add the complexity, you are stuck with that processor speed.

So, that's probably more of a question mark for our codes than it is (audio interference), although we're trying to catch up.

MEMBER MARCH-LEUBA: I'm quite sure that we don't run off just into the box corner. So, you're just kind of updating, because it doesn't run.

MR. BAJOREK: It used to run on a vax.

MEMBER MARTIN: Okay, and one question on capability. Among all those codes you have there on that slide, at least (audio interference), I probed a few years ago, when it came to, say, a balanced plan modeling, really triple machinery modeling, the codes -- maybe SAM has something now.

Obviously, a code like TRACE has the old simpler models. Where do we stand in code capability on modeling balance of plant, triple machinery?

MR. BAJOREK: There's some of that in

SAM.

MEMBER MARTIN: It's coming.

(Simultaneous speaking.)

MR. BAJOREK: Various types of heat exchangers, pumps. It probably needs a little work on the valves. Okay?

MEMBER MARTIN: Urban model compressor --

(Simultaneous speaking.)

MR. BAJOREK: No. That's not in there. That's another reason why we have TRACE part of the mix right now.

MEMBER MARTIN: Right.

MR. BAJOREK: So, as we do the secondary tertiary systems, or if we have hydrogen production, and some of the other ideas on distributing the heat, well, we can do that with a TRACE --

(Simultaneous speaking.)

MEMBER MARTIN: We can do it with, say, like a gas? So, if you had a gas cycle so we can run air, or --

MR. BAJOREK: Actually, we have some updates for super-critical CO_2 systems to put in the TRACE. They haven't been tested. I wouldn't try to run out and use them.

But we have properties for sodium, a transfer for sodium, we have some lead, we have a variety of molten salt properties within TRACE, helium, a couple of other gases.

MEMBER MARTIN: Okay.

MR. BAJOREK: They're there. So, as we deal with those other systems, if it's a secondary salt system and we have to deal with valves and pumps and things, we can do that.

And whether the pump is the same type of -- and for molten salt, I really don't know. But that's been one of the ideas about this whole thing, is that we'll use TRACE for those other systems, where you can model it in a simplistic fashion, but you want to see the effects of tertiary systems that can also fail, or have glitches, as we understand these plants.

And I mentioned the importance here, and the complexity comes in because so many of these systems are coupled neutronically, done fluid-wise, where the fast reacts with the tensor mechanic. That's one of the newer twists with these non-LWRs and why we went in that direction.

Verification and validation. As I

mentioned earlier, that was a question mark five years ago.

What's out there? How much is out there? And is it really sufficient to understand what's going on in these technologies in these codes?

So, we put together a V&V Report. It says draft on there. I think it's still on there. It's draft in the sense that we intend to update this.

As more assessment is done, as other verification, as we get more PIRTs, and things like that, we're going to build those into the document.

I'll go through some of the contents to describe it a little bit better. But the idea was, okay, let's (audio interference), available PIRTs, and there have been two or three additional ones developed since we did volume 1.

We did a couple of our own for molten salts. There's a new one by Westinghouse on event sheet. So, I wanted to make it easy for others who are looking at this to see what's really out there.

Verification standards, that was a

question mark. How do you verify things? And how do you make sure that if you add an update, you haven't broken something else. So, we wanted to at least pay some attention to that.

And then go by major technology. What assessment has been done to collect all that? And at least get citations on where it's at. And as I mentioned earlier, it's spread around.

If you're developing a code, an Idaho, if I'm doing GRIFFIN, I'm worried about GRIFFIN and I'm assessing that. (Audio interference) SAM, I'm doing SAM.

I'm doing a whole system, and it may not pay much attention to that. So, I'm looking for things that have these coupled assessments along with those that help benchmark individual code.

At the end of the day, I think it helps us identify what assessments are out there, what's been done, and what work do we still need to do?

Okay, there's still stuff out there that needs to be shored up. And in doing this, I kept getting confused on what is HTR, HTTR, HTTF, HTRPROTEUS, and everything else that starts with

HT. So, we wanted to put a quick reference on test facilities and benchmarks, to just identify what each of those can do.

The contents we go through, and much like volume 1, we describe the BlueCRAB codes, make some reference to our integrating scale shift and incorporating sockeye, a little update on that, PIRTs and scenarios.

And then we go by technology, with some separation for neutronics and individual components -- heat pipes, pumps, what I call local phenomena, because what goes on in an upper or lower plenum has importance to several systems.

So, we separated that down there and identified the various tests and code-to-code benchmarks that have been performed.

Ms. Bier, real quick, when it comes to verification of these codes, you're really relying on DOE and their own methods.

My experience with them interfacing with industry, obviously, a big selling point was that they were coming in within QA-1 approaches to everything, and that industry would never need, say, source code to verify.

Now, we're also getting into a time where export control becomes a particular challenge. The days of our previous light-water reactor vendors having access to all the codes and doing everything is probably going away.

That, maybe from an industry standpoint, should be a good thing if they are no longer responsible for the first V.

Where do we see the evolution, and who's responsible for what, given the realities of doing nuclear assimilation?

MEMBER MARTIN: That's a real good question. Because I think the issue that could arise is if I'm an applicant and I've done an analysis, and somebody over here has astutely identified code error in one of those means tools, or whatever tool I'm using, who owns it?

Okay? Well, I think the NRC will ultimately go back to the applicant. It's your analysis. You own it. Okay?

How you resolve it by working with whoever developed the code, is going to have to be a problem between those.

But we all want the codes to be accurate. And we're going to see the same problem.

If we see a code error in TRACE, well, we go and fix it. If there's any that messes up an analysis, well, we report that.

If it's out there when we're going to have to work with DOE to get it resolved, and it we made conclusions that were erroneous because of that, we're going to have to deal with it. So, it'll be a more complex situation.

MR. BAJOREK: Right. And certainly less agile. When the companies that are advancing these things don't have some control over the source code, it will invariably create delays and heartaches, and there will be errors.

MEMBER MARTIN: Yeah, you have to fix them. The other issue that I don't know if some of the applicants have thought about, is that if a public code is being used to analyze their design, anybody can go get that and analyze their design.

One of the reasons you bring that code in-house as a vendor, is to put your own stamp on it, you own it, but now it becomes proprietary. And you don't have to worry about some job shop, or going out there and doing your reload analysis.

MR. BAJOREK: I don't know. It's just different.

MEMBER MARTIN: It's just different.

MR. BAJOREK: Faustian deal, you know, you may get some benefit that you don't have as much responsibility. But when there's a problem, then you could be stopped.

MEMBER MARTIN: Okay, the content of the V&V Report, I just put a snippet of one of the tables over here.

And what we try to do is go through, identify the test, the test facility, TFK&M, whether it's certain fluids, kinetics, fuels, or a mechanic's -- what got exercised, what codes got involved in that particular analysis, whether it was a separate effects test or integral effects test, is more separate effects test than what's indicated here.

What design type it likely pertains to, and then the validation references that are on there. And then the references are not consecutive in order, because they all came in at different times, and they couldn't get them done in consecutive order.

But we wanted to identify some by placeholders. As you see down for HTR10, there's some analysis that planned using SAM. That work, when it's done, will fill in that missing reference.

So, you can see what's there. The yellow highlights, that's the stuff that's out there that people would like to do, or intend to do, or possibly could do.

There are some, like THTR300, which is a thorium high-temperature reactor. Now, everyone may look at that and say that doesn't really quite have the applicability. It's not worth the effort to do that assessment.

So, that might be dropped off the list. But the idea is, let's put all the tests that we know about on here and use that as the potential assessment date.

The things highlighted in blue are things that have gone into the virtual test bed. These are models which have been developed for, in this case, the HTTR and the HTTF. They're available.

They're publicly available, as I understand. They're good building blocks. Don't compare to data, but they're important to keep track of.

Some overall comments on the report. Contrary to maybe what some of us thought five or six years ago, there has been assessment completed for all these technologies. There's something else out there.

You do get a sense of maturity by looking at the amount of work that's been done. Not that you want to judge the assessment or maturity based on pounds of paper that you've been produced, but as you look as gas-cooled systems, sodium liquid metal systems, there's been quite a bit of work.

They've received a lot of attention. There's a sizeable database that has gone into a lot of the assessment.

On the other hand, if you start to look at molten fuel salts, now you start to see the list get much shorter, and a high dependence on the MSRE.

Everyone in that last column says, oh, MSREs are validation. That's how we're going to assess it.

Well, there's a couple of things

there. MSRE's a ten-megawatt thermal reactor. And some of those systems are a couple of hundred megawatts. Much, much larger.

Some are loop systems, some are pool-based systems. So, now the challenge eventually is going to be, can I take MSRE and its five and its constituent makeup, and scale that to these other designs out there?

There's not a whole lot else out there to go on. And so, that's a question. You know that some of the applicants are doing their own work, their own tests. We haven't seen that yet, but that would be needed to possibly mitigate that possible concern.

MEMBER PETTI: MSRE didn't have power conversion either, right?

MR. BAJOREK: No, it just dumped it out to the parking lot, yeah.

MEMBER PETTI: So, that's a big difference.

MR. BAJOREK: Yeah, it's a big difference. The enrichment of that one is probably different from what some of the applicants are thinking about. Things slide, but -- I get fascinated with the thermal physical properties.

A lot of these are eutectics. They're no longer eutectics once you've done fission products in there. And that creates another uncertain with this cost-based thermal cod activity and corrosion products, which we're not dealing with, but it's something else that --

(Simultaneous speaking.)

CHAIR KIRCHNER: Is there a way for you to split that third bullet, Steve? I mean, the word fuel -- there's molten salt reactors, and then there's molten fuel salt reactors.

And that's an order of magnitude, more complexity in terms of coupling, and some of the issues you pointed out.

Once you add fission products into the salt, that changes properties. It introduces corrosion, it introduces complexity. Considerable complexity, versus just molten salt, which is challenging enough, or code sets.

VICE CHAIR HALNON: Are you going to have a relationship with, like, ACU, to get some of these questions answered? Or even maybe sanction some tests that might help answer the questions?

MR. BAJOREK: Well, that's probably going to drop down to the final bullet down here, that when it comes to assessment, our codes, probably their codes as well, for microreactors, and including the fuel salt, the ACU design, that's going to depend on these prototypes. Okay?

There's one being a bullet, there's a couple of others out there. Illinois and ACU are proposing basically research and test reactors.

That's really where the assessment data's going to have to come from.

VICE CHAIR HALNON: I see a commercial collision here, with the scientific community needing this information. So, yeah, that relationship's going to be dicey at best.

MR. BAJOREK: Yeah, that's one of the things that at least stood out to me. As we look at microreactors, the fuel salt, the assessment base is weak. And it has to be augmented, either by applicant work, prototypes, we see the MARVEL Reactor going up at Idaho, that's going to be helpful and useful.

But that's sort of a microreactor that's not like some of the other ones. It's an (audio interference) instead of heat pipe.

So, anyway, my hopes with this V&V Report is that helps identify technical and knowledge gaps, and assessment gaps that we're going to have to address over the next several years.

MEMBER MARCH-LEUBA: Okay, this is a different topic. I count, like, over 100 slides left. It's already 10:00, and I start moving.

MR. BAJOREK: Okay. I can move it --

MEMBER MARCH-LEUBA: It's not a fault. I just, I must confess.

MR. BAJOREK: No, no, no. I'm usually overly optimistic on how much I can cover in time.

I have examples in here for all of our reference models. I'm only going to do one or two of those. Okay? Because it'll get to be repetitive.

But the reference models, these are, for the most part, generic public information-based models of something that is fairly close to what we think the applicant's going to do.

The scenarios are things that we think are going to be part of their design basis. We don't know exactly what those are, but we're basing that on the work in the past.

And the main benefit, apart from educating staff in how these codes work, how the technology kind of works, is we want to identify deficiencies in the codes now.

We don't want to wait until the application comes in, and then realize that, oh, we don't have a mechanism of chasing neutron precursors to the fuel salt. Or, gee, you know we put in a solidification model.

Or, we have a complex structure and we can't have independent channels flying through. Just one of the things.

So, we found a number of those things. But the idea is, we want to develop those reference models, train the staff, and as I tell the people we work with, get it in-house and break it. On some sensitivities.

And if there's problems, it doesn't converge, it fails, there's something -- a capability that's not there, then we go back and we talk to NEAMS and say, let's get that done now. Because we cannot hold up the review with the intense pressures that we're going to see to get these done in a short period of time. They're designs that we've never done before.

So, I'll start off with what would have been more of a conclusion. And I want to just point out some of the things that we've done.

On the left-hand side you can see the six referenced designs that we've been working with primarily. And over the course of the last five years, we've made some big steps forward.

As we look at the gas-cooled pebble bed, or the molten salt mold pebble bed designs, we have methods now for doing pebble tracking, getting to equilibrium core, doing a great job predicting the radial and axial --

(Audio interference.)

CHAIR KIRCHNER: Can I remind everyone online, please mute your microphones. Thank you.

MR. BAJOREK: We started off by looking at the core and then moving outward, modeling the vessel, and then we're now at the point where we've added on reactor cavity cooling systems, secondary systems, DRACS systems -- most of the models have that.

A couple of them, there are some

improvements. We did some real simplifications on the loops. We can build that in, and in a couple of cases the RCCS is single-phase, but we'll merge TRACE in to do a little bit more funnel when two-phase flow occurs.

We can get a fair amount of detail and still run it quickly. The ABTR Model for sodium fast reactor, we model all 61 channels individually. Sixty fuel channels, a bypass channel, and couple that with the tensor mechanics to look at the plate expansion and the fuel expansion, which are your major contributions to negative reactivity.

The MSRE, everyone's go-to model for a fuel salt, we now have the thermal fluids code and the neutronics code coordinating with tracking the precursors. That's very important. We may not get to it, but you have a loss of flow. Those long-lived precursors now represent a positive reactivity in the core, as opposed to losing their neutron in the upper plenum somewhere out in the loops. But we can do that now.

Heat pipes, five years ago, was just a superconducting piece of metal. For a lot of part, that actually does a reasonably good job.

But we have the simplified model in SAM that can handle that. We're merging in Sockeye for more transient types of situations.

And as I mentioned, there are other models that are available to us from the GCR that we've exercised. You see some of those down below.

But one I'll use as kind of an example on the approach for this is the gas-cooled pebble bed reference plan.

We selected the HTR-PM, which we think is a reasonably good representative of what we think X-Energy's going to come in with the X-100.

HTR-PM, there's two of these operating in China. They're both at 250 megawatts thermal. X-100, based on our public information, 200 megawatts.

If I didn't put the label on here, I'd forget which is one or the other. But as you go through the system, the geometry, the rank and cycle that is built into the system, flow arrangement upcomer, upper plenum through the pebble bed, there's a lot of similarities. So, I don't think we could get too much closer on public information on this.

In the interest of time, I won't talk about all of the details. HTR-PM has about 420,000 pebbles in there.

We took that, we meshed the core, the reflector, the vessel, we've add-used SAM and GRIFFIN to provide a 2D porous media within the core and the vessel, one-dimensional representation for the loops, a simple air-cooled RCCS.

It's just basically, get the energy out. We're not trying to do a detailed representation, because we don't know what the model looks like for the X-100.

GRIFFIN was exercised for getting the equilibrium core, doing depletion. There were some questions on how long it'll actually take to get to an equilibrium core. Some interesting questions on that.

It gives us that axial and radial power distributions, which you see here with the power high in the core, your higher temperatures down lower as you go to the exit chute, salt and the fluid temperatures fairly close together.

And the right-hand side, cut off on

that slide, is the fluid velocity accelerates as it goes through.

MEMBER MARTIN: Steve, a clarification. You don't need GRIFFIN necessarily, to do something similar. You can still use SAM in a traditional sense, where you have a point kinetics model in it, and set up each structure.

And there's a fitter model that I'm familiar with, without bringing in the more complexity of GRIFFIN.

MR. BAJOREK: This is a case where we decided to add some complexity now, in the hopes that when we use it with GRIFFIN and we get our parameters for point kinetics, maybe we can do this simply with point kinetics as we go on.

But we're never going to know whether that's sufficient unless we look at something that's a little bit more complex right now.

So, my neutronics experts say, oh, don't go straight to point kinetics. Let's explore this.

It's still my hope that we kind of show that we can go ahead and take a more simplistic approach, but that's one of the things, we want to sort that out now.

I don't want to have to face that question a year, year-and-a-half, into a review, and have to deal with NRR and say, hey, the applicant's using point kinetics, is that good enough?

MEMBER MARTIN: Yeah. My real question was, the old way of being able to do the simple models does exist.

MR. BAJOREK: Yeah.

MEMBER MARTIN: But you've brought in the complexity because you can.

MR. BAJOREK: Yeah. Yeah, we'll do it now, and if we can drop back to a simpler approach, by all means do that. Especially if we get to the uncertain stuff.

MEMBER PETTI: But in salt systems, I don't know if the point connect's going to do it.

MR. BAJOREK: No. There's some systems, it may be feasible for others.

MEMBER PETTI: Right. There's not going to be a one-answer, one-size-fits-all, so to speak.

MR. BAJOREK: Getting codes coupled has its own issues. And it's nice to test it out in different ways. Again, let's break it today, so we don't have to deal with the damage out in licensing space.

So, anyway, we've done the steady state transience and flow, we've done an overcooling transient to give us a reactivity insertion by -- I think we fail a bypass valve and we get cold helium into the system, so we see the response of that -- a PLOFC, a DLOFC.

Anticipating a risk-informed world, we're going to be looking at a small leak, or a small LOCA, from the system. See how that progresses.

(Simultaneous speaking.)

MEMBER MARTIN: What code grade is the applicant using? Or do they have different codes in the name suite that they're using?

MR. BAJOREK: There is a mix. There are some that are using some of the names tools. There are others that are using their own.

For the gas-cooled reactor, the latest I saw from one of the applicants, they were using a CFD code for the thermal fluids, and I think it was VSOP for the kinetics, which is my understanding, might have been a South-African vintage code.

MEMBER PETTI: It's a German code.

MR. BAJOREK: German? Okay.

MEMBER PETTI: Goes all the way back to the Germans.

MR. BAJOREK: Okay.

MEMBER PETTI: Called Very Sophisticated Old Program.

(Simultaneous speaking.)

MEMBER MARTIN: So, how independent do you think you have to be if somebody came in with an analysis using the name suite? Do you consider yourself not adequately independent, or --

MR. BAJOREK: I think we'd be okay. Because one of the things that you find in using the codes, if you give five different people the same codes, all equally qualified, have them go do an analysis, you'll get five different answers.

The biggest uncertainty may be the user effect. So, I think we're reasonably safe by doing our own independent analysis, making our own assumptions.

Because for something like this, we

may say, we think the right thing to do is to use KTA correlation for pressure drop and you nodalize it in a certain way.

Applicant may choose something else. And they may model things in a different fashion, make different assumptions. So, I think we're reasonably good. Of course, we'd be much safer if they did something different. Anyway, I'll just do this one quick.

MEMBER MARTIN: Is that something you would assess if somebody came in with the same code suite you're using? You'd then have to take another fresh look at how independent you are?

MR. BAJOREK: Yeah, I think we'd have to.

MEMBER MARTIN: Okay, thank you.

MR. BAJOREK: And as I go through each of these reference models, we'll run through the transient, in this case a pressurized loss of forced cooling.

This just shows the pebble temperatures. Their greatest temperature at the start of the transient near the bottom of the core as the flow stagnates, you don't get much recirculation.

The design has the hotspot up here and the cold spot down here, so the loops kind of shut off flow through those.

You get some recirculation in the vessel. You see a lot up in the upper plenum, a little bit down through our -- and then the upcomer. Temperatures become hotter at the top of the core. You see the reflector heating up, with the energy eventually being taken away by the reactor cavity cooling system.

I don't have my notes on here right now, but the transient takes a number of hours. I think it takes on the order of ten, twelve hours, before the decay heat in the system is completely removed by the reactor cavity cooling system. So, it's very slow transient, but we can run these in reasonable time.

MEMBER PETTI: So, this is one of the cases where what you worry about here is not so much the core -- people have done these calculations forever -- it's the vessel, and whether you can fail metallics that are at the top.

And so, sometimes it's going to take some thinking, as the saying goes, out of the box, so you're going to have enough fidelity in places that you wouldn't necessarily have to have fidelity, in other systems.

MR. BAJOREK: When we get to next steps, and explicitly, one of the things that we're trying to do with this model in particular, is do a better job on the upper plenum and the lower plenum.

Because I agree entirely that fuel temperatures up to I think 1,200 or 1,300, big deal. But the vessel temperature in the upper head potentially where you have the weld and the cross-connect pipe, those are places that we want to --

MEMBER PETTI: Yeah.

MR. BAJOREK: So, we're moving from the core on out. One of our emphasis on the work right now is doing a better job on getting vessel temperatures and temperatures in locations on heat exchangers that -- no-never-minds and light-water reactor space, but they're not going to be that way now. So, thank you for that --

(Simultaneous speaking.)

CHAIR KIRCHNER: This reference model has an active -- the KU removal system with a cavity?

MR. BAJOREK: As a reactor cavity cooling system?

CHAIR KIRCHNER: Yeah, operable to this transient?

MR. BAJOREK: Yes, yes. Yeah, in the particular transient the rods was a SCRAM. The rods dropped, goes into decay heat, and that has to be removed by the reactor cavity cooling system.

I don't want to abuse my time period unless I can get away with it. But talk just briefly about the ABTR. That's another one we've been putting a lot of work into, because we see the gas-cooled pebble bed sodium-fast reactor, and the pebble bed molten salt is being kind of the leaders in where they're at and coming through licensing.

As I mentioned, we can model, and we do a rather sophisticated model of the core, model all the different types of assemblies in the 61-channel representation.

We've modeled the DRACS System, simplistic in the upper and lower plenum. And that's another place that we need to do a better job in the future.

The interesting thing about the sodium fast reactor is probably the reactor dynamics. Okay? We use GRIFFIN to get the reactivity coefficients for Doppler, axial fuel expansion, sodium temperature and density, and HE is the radial thermal expansion of that plate.

And that's where BISON came into play for us. The model that -- kind of the complex plate. We had to do it in two different regions -- that outer wing, darker in the thinner sections in the middle, but as we run the unprotected loss of flow, the sodium -- if I have that on here.

Okay, we run that one, we lose power to both the pumps, the flow decreases, we drop it down to like one percent. We didn't have details on what the pump was like.

But you very quickly start seeing sodium heat up that lower support plate expand, you increase the leakage, and that's your major negative reactivity component.

And power decreases and the transient goes off.

MEMBER PETTI: So, Steve, some of

these designs, they're allowed to just dilate the way they're going to do it. Others, they want to constrain the core. Okay?

MR. BAJOREK: Yeah.

MEMBER PETTI: And so, that capability's got to be really critical.

MR. BAJOREK: Yeah.

MEMBER PETTI: Because if you're trying to optimize things so you restrain stuff not to move, that's a pretty complicated problem.

MR. BAJOREK: It's a tough one. And it's one of the areas with the MOOSE framework I think it's going to be very beneficial to. That's because -- we didn't do it here, but if we had to look at flowering of this, you can do it. I don't necessarily think it's easy, but --

MEMBER PETTI: And I can tell you, I mean, we're reviewing some documents. And the flowering, and then preventing the straining --

(Simultaneous speaking.)

MR. BAJOREK: You've got the core restraint.

MEMBER PETTI: I mean, that's complex.

MR. BAJOREK: I'm going to jump ahead here. We can always to back. Yeah, I'll just mention the MSRE. That's been our fuel salt.

We've modeled the porous media approach within the core. We've added the loops, simple heat exchanger, but the important thing there is that we're able to identify and track the various neutron precursors to the system. The short-lived ones are on the right, the long-lived ones are over on the left.

As you can see at Steady State, a lot of those long-lived precursors release their neutron as you're either up at the top of the core, or you're getting out into the system.

That makes the transient very interesting. One, we do have some data on there. We've got favorable comparisons to the data that's available for like a pump startup and coast-down.

For the unprotected loss of flow at zero and full power, when you lose that flow, now those neutron precursors, they stay in the middle of the core.

That's a positive reactivity. Okay? But the Doppler and the fuel salt density, which decreases, those are negative. Those mitigate that situation for the MSRE.

Now, would that behave the same way for a natural circulation system? Well, this is where we think we can deal with it. But until we really see those systems --

CHAIR KIRCHNER: Yeah, the actual system is going be very important. Fluid velocity, among other things. It's a lot more complicated as this tail goes up.

MR. BAJOREK: Yeah.

MEMBER PETTI: And vacate that vast salt systems even more. Higher providence that you could void -- I mean, I'm not convinced that the delayed neutron, you know it's even controllable. I mean, there's all sorts of issues. That's one of the ones -- box out there.

MR. BAJOREK: As I say, we get the models, we bring them in-house, here's a very simple one, one D-core, the model, the MSRE, and we had a staff member take this, do a better job, do a better model on the intermediate heat exchanger and the pump, the secondary system, go break it.

And he came back the next day and found a way to break it.

The MSRE actually, the elevation

between the heat source and the heat exchanger isn't all that much. You change that elevation a little bit, you completely defeat the natural circulation within the system.

And so, he did that. And by changing that elevation just by its (audio interference) showed that, yeah, you get temperatures in the core you won't want. So, that was that.

We've done work with the microreactors. And we've looked at two different flavors.

When we took a look at -- it's a design by INL and then LANL; I think they played a role in it too, the modified special purpose reactor -- but we modified it and the way we came up with our own microreactor. Because, one, we wanted to use metallic fuel, not oxide, and we wanted it to be a fast reactor with thick heat pipes, large diameter heat pipes, as opposed to the thousand smaller ones that they had in the special purpose reactor.

And we've also, we have a model that's being developed right now, we've got a little bit of results for, an eVinci-like, both based on public information, information that we have out there.

But they've been very useful. Because we wanted to look at things like single heat pipe failure. Question was like, if I fail a heat pipe, or heat pipes -- nobody said only one can fail if you're not monitoring them -- well, we've set up a model, we failed one of the heat pipes.

The fortunate thing that we found is that when that heat pipe fails, the temperature in the core heats up just a little bit. That reduces your power of the core just a small amount.

In the vicinity of that failed heat pipe, its temperature increases dramatically. One surrounding it increased. They pick up the load of that failed heat pipe.

We did not see temperatures in this particular scenario, where that failed heat pipe would cause you to cascade, or do any others. Yeah, this is just the example that we've done. Change the design, something would have to be looked at. We think we're prepared to do that. And hopefully, you see that same type of margin, as you would for a loss of heat sync. Okay?

The heat pipes go through the

condenser and they give up their energy to an external cycle or something else.

Well, if that were to completely fail, your heat pipe removal basically goes to zero. Anything that's circulating when the heat pipe stops, you start to increase the temperatures in the core very quickly. But because of the strong Doppler, that decreases the power.

The other important thing with this one is we had thermal-mechanical expansion, because it was a fast reactor.

As this one heated up, you also had an increased amount of leakage from the core. That also helped shut down the reactor and mitigate getting to exceedingly high temperatures. So, anyway, that's the capability that we have out there.

We and the other volumes are taking what might be called a multi-phased approach, probably more so in volume 1 in the other ones. Some of the details matter.

We want to make sure that we first exercise the codes. If we find problems, let's get them fixed, and then let's gradually add complications to the model, make it more detailed, model things that we hadn't in the original model, to get it closer and closer to a specific application.

Stage three, that's when we get proprietary information. We can at least take it as far as we can right now. The transience, the modeling that makes it look like the applicant design, but to go further, we need the applicant to come in and give us good information.

A couple of them have been very good. We're working on an eVinci model now that's going to be eventually close to what we think the applicant's coming into, the publicly available information.

We ran that one and looked at that one, and we said, yeah, well now we know what changing it to be. You can see some of the issues though that are corrected and where we're going.

MEMBER MARTIN: What you describe sounds still very manually intensive. Is there thought of automation in some way, to go from reference plan to something that more design-specific?

Automation isn't always attractive

from a funding standpoint, but everyone -- all the applicants are doing it. Because, of course, all the applicants are doing it. It becomes a competitive sort of thing.

Because they're agile. I mean, isn't there some onus on the Agency to be agile with doing these?

MR. BAJOREK: One of the things that I hope we can take advantage of, is when we set up these models using the names codes, there's a certain architecture to them.

You define the kernels. Basically, the partial difference in equations, how it's set up. And then the mesh is developed elsewhere.

What I'm hoping we see is the design changes, so we can modify the mesh. And people who are good with that seem to be able to knock it off in a day or two, change the mesh, but the rest of the model may not have to change.

So, hopefully, if there's not too much deviation, we can do it quickly. Completely automating? Maybe someday.

MEMBER MARTIN: Not a priority?

MR. BAJOREK: I can't think of how to do it right now.

MEMBER MARTIN: Well, maybe to the point of my earlier question on verification role, verification of elevation, these are effects, right?

We have guidelines, right? And of course, we've gone into all the gory detail they did at the volume, the paper.

You know, there's a lot of attention that goes into guidance on how you model. It's got to be different now. I mean, because you don't control meshing -- I mean, there's probably some control over the density of meshes and what have you -- but it's just not the same focus that you would have with a finite volume approach to the code.

MR. BAJOREK: You're right, we have user guidelines for TRACE.

MEMBER MARTIN: Okay.

MR. BAJOREK: And I would see someday as we evolve to systems that are becoming more stable -- not stable, but I mean we know what we're really getting into. And we developed guidelines for probably each of the applications.

I don't know if you can come up with a generic set of guidelines and how we're going to do a sodium fast reactor to a microreactor.

MEMBER MARTIN: Well, the developers of the codes at the different labs, I mean, in their documentation are they capturing kind of these guidelines?

Historically, they would, harkening back to my work on the Real Five development team 35 years ago.

But is the documentation complete to that extent? To not just the code structure and models of correlation, but also, and beyond the development assessment that actually includes user guidelines?

MR. BAJOREK: Not to the extent of guidelines. The input manuals that I've gone through for the names codes, they give you the flexibility. They don't restrict you in a certain way.

In TRACE, we write the guidelines more, and the models and correlations are locked up. You really can't go in there and decide to use a different one.

MEMBER MARTIN: I mean, TRACE has got to have the flags, like RELAP-S.

MR. BAJOREK: Yeah. Yeah, they're

saying like that. But we say, hey, if you're doing a large grade local, these flags need to be on. You need to model in a certain way.

With the names tools right now, I have not seen that being defined. It's probably more in the developer's head right now than it is on the developer's paper.

MEMBER MARTIN: Right, right. Of course, you meet with those folks, and there's feedback, right? So, I would certainly believe that would be the kind of feedback would come from Agency back to DOE.

MR. BAJOREK: Our next steps. We're still refining the reference models. A question right now is, if we have an asymmetric event, to what extent do we need to put in multidimensional models in the core to look at some of that? So, we're determined to investigate that.

We're going to be looking at whether we can incorporate PRONGHORN to give us some of that detail. Or, we can just stay with the 2D RZ formulation that we use with SAM.

We'll test that out now. We've largely ignored secondary group models. They're there, but we can improve on some of that capability.

And the RCCS, there's at least one applicant out there that wants to flood those tubes with water. We think they may even get a quench rod in there. So, that's an opportunity for TRACE to be able to look at the operation in a two-phased environment.

As I mentioned, one of the things we would like to do is to incorporate better methods for doing sensitivities and certain methods in here. We've talked about that but we have to get that into our van as well.

Validation. As we go through the V&V Report, you can see where some of the gaps are. And we'll talk to DOE and say, we need to accelerate the pace of what's going on here. Probably heat pipes is the main one right now.

Once scoured again, and look for places where the database is clearly weak and point that out. And hopefully, that can be corrected by the time an applicant comes in. That's up to DOE and the applicants.

And like I say, as we get better information from the applicant, we'll build that in. So, hopefully, when we get into the review stage, we're ready to go. And hopefully, we can demonstrate that they believe what the applicants are telling us, that there's a lot of margin here in that design, and that assists the review in moving forward.

MEMBER PETTI: Steve, just a quick question. We haven't talked about ingress events. So, water in the gas reactor got to be a deep inside the design basis, right? Because they're going to have a heat exchange.

And then in some of the micros, yeah, ingress, depending on what the configuration looks like. Whereas, in some of the others, they'll say it was beyond design basis. So, that, I think, MELCOR. But there may be some that it's going to be --

MR. BAJOREK: Yeah, we talked about that when we did the DLOFC for the pebble bed. The boundary conditions were such that we were getting error and kind of objected to that.

But it's probably one of those areas where there are some transience where I think BlueCRAB can do a better job.

But there's others where I think you need to go to MELCOR. And I think air and water ingress are those, as you get --

MEMBER PETTI: Okay, so you just used MELCOR, even though it's a deep end (audio interference).

MR. BAJOREK: Yeah, it might be for the DLOFC. You get a lot of water ingress, or graphite dust and all those entertaining things. That's probably more of an MELCOR. We might be using BlueCRAB then to say, hey, here's what we think the radial power distributions are. This is a way of --

(Simultaneous speaking.)

MEMBER PETTI: It's just that on the steam generator-2 failure, you can get a reactivity. So, like, whether MELCOR could handle that, versus you guys never set up.

MR. BAJOREK: Because I've used my time, but --

MEMBER MARTIN: At least one last question for me. Of course, the MDEP process -- 30 steps, 40 steps, or whatever it is; one of them relates to code scaling, scalability -- the practice of scaling has traditionally been more of a specialist, oftentimes relying on people that come from the testing world or what have you.

When it was hot and heavy 25, 30 years ago, you had a lot of experts in that area -- there are fewer and fewer now -- which begs maybe some attention to the code development, and maybe the kind of figures of error that we can draw from it.

And I know the answer to this ahead of time. Has anything been done really to facilitate that aspect of the MDEP process, to help really practitioners to understand scaling and scaling phenomena, similarity criteria --

MR. BAJOREK: In integral systems, I have not seen much of that. There has been some really nice work done by Peterson to scale surrogate fluid to molten salt. So, you can use water in place of the high-temperature salt. And I think there's also been some systems consideration in that work.

There actually has been work done on heat pipes, as one of our questions was, well, we see some of the applicants with very large diameter heat pipes. Very long heat pipes. Twenty, 24-feet, something like that.

How does that scale to the pencildiameter heat pipes that are used in satellites, laptops, and stuff like that?

And I saw a nice scaling report where, yeah, you can take some of those and scale those up to a larger diameter.

MEMBER MARTIN: So, for like preparatory analysis efforts where you see the role of these tools to support the valuation of scaling, I mean, I don't know if it's 0 and one-off kind of effort analysis with a slightly different focus?

MR. BAJOREK: Well, I guess when you say scaling though, I've always interpreted that as the scaling of the experimental facility to the full scale prototype.

MEMBER MARTIN: Right, right. But there's multiple --

MR. BAJOREK: Codes don't --

(Simultaneous speaking.)

MEMBER MARTIN: Code scaling enrollment. I think of applications I've been involved with, where we would use the codes to evaluate non-dimensional parameters in a dynamic sense. You would oftentimes be looking at distortion over time come into play there.

We would make complex control systems that would otherwise draw out that kind of information. Of course, we'd do shorter models on the side to complement that.

I mean, it was its own industry, if you had that resource. But because it was such a unique competency, dropping that into the NRC, it's dropping a rock in there. Because the building would have to be developed. I mean, it's not easy.

And I do feel like that aspect, MDEP, is not getting the review that it was intended 30 years ago.

MR. BAJOREK: No, you're right. I mean, I think it was when MDEP and CSAU is when they got --

(Simultaneous speaking.)

MEMBER MARTIN: Right. I mean, but the test scaling has been around since forever -- 70's, and ISHI, and those sort of methods.

But with regard to the integrity of codes, a valuator, from a scaling perspective, which was a popular topic -- it certainly seems diminished -- I do think there's opportunity in co-development to tackle that.

But again, it depends on the kind of questions that come from ACRS members, the kind of questions that come from the staff here, and what have you, on what attentions to get.

And I don't know if we're losing it, if we worry about losing it.

MR. BAJOREK: I hope we keep it on the table. Because I think it's going to be at least there as a way of showing that the data that's been produced is truly applicable to the system.

MEMBER MARTIN: Sure.

MR. BAJOREK: We're not ready to do any kind of code simulation where you change that yet. We're happy to get the code to run.

MEMBER MARTIN: It's still hard. That's basically -- it's still hard.

MS. WEBBER: If I could add just add one comment. The big push over the last seven years is to get capability. And a lot of the questions I really have appreciated, and comments I've appreciated.

But that takes the capability to the next level. And that will happen over time and with resources.

So, it's not that we don't appreciate

the comments. It's just we're trying to build a basic capability to look at what's in front of us now, and then to be able to address these much more dynamic, complicated situations, as we move in the future.

MR. BAJOREK: I think our job's being patient.

MS. WEBBER: Yeah.

MR. BAJOREK: To throw it right back into --

MS. WEBBER: But we do appreciate the comments and insights.

MR. BAJOREK: Yeah, we're still on the first couple of miles of a marathon, when it comes to really understanding and licensing some of these designs. And I think for all of the codes you're going to hear today, we've made a lot of progress over the last five years.

When it comes to the BlueCRAB, I think we're about ready for doing independent analysis. Give us the design, I think we can tackle it.

We've got reference plants for a number of these designs, especially the near-term guys that are out there, and that's helped us with our understanding.

Looking at V&V. Okay, that's why I put the V&V Report together. To see where we're at, what's mature, what needs to be done. So, that's put us along that path.

And I'd like to say that BlueCRAB is tentatively ready for independent analysis. We've dealt with the known unknowns, to the extent that we can -- the database available -- but there's going to be those unknown unknowns.

We don't know what that design is. And there will be work that we're going to have to deal with, whether it's scaling, whether it's a mesh sensitivity, how you model a certain grease plug or DRACS system, things like that. Those questions are going to be out there.

But I think to the extent that we could have done so in the last five years, I think we're in a pretty good place right now.

But I'd really like to thank you for your attention, your questions.

MEMBER MARTIN: All right. Yeah, we've kind of come to the conclusion of this first presentation of several today. Last questions from the members? Hearing None, do we just go to recess?

CHAIR KIRCHNER: It's a good time to take a break, right?

MEMBER MARTIN: Yep.

CHAIR KIRCHNER: Let us take fifteen minutes and come back at 10:45. With that, we'll take a short recess break here. Thank you.

(Whereupon, the above-entitled matter went off the record at 10:30 a.m. and resumed at 10:46 a.m.)

MEMBER MARTIN: Rejoining our meeting here on the non-lightwater reactor code development. We've heard from Steve Bajorek with the volume one, we're moving into the subject of fuel performance analysis. Kim, did you want to introduce who you have here for us?

MS. WEBBER: So I'm behind you, James Corson is going to call in. And James Corson is a senior reactor systems engineer in my division, he reports to Hossein Esmaili in the Fuel and Source Term Code Development Branch, he's going to talk on the fuel performance volume two progress. So James, are you online?

MR. CORSON: Yes, I'm here. Can you hear me?

MS. WEBBER: Yes, we can. Take it away.

MR. CORSON: Okay. Good morning, everyone. Unfortunately, I couldn't be there, I'm actually on travel this week and I've had other meetings that have kept me away from most of this morning's session. But I'm happy to talk to you now, about our fuel performance analysis for non-LWRs.

So, as you know by now, we had written a plan to look at fuel performance analysis for non-LWRs, dating back to 2019. So, the whole goal of this plan, and fuel preference in general, is to understand the thermal mechanical nuclear fuel behavior during normal operations, anticipated operational occurrences, and accident conditions. So, the goal of our tools is to be able to provide insights for developing regulatory guidance or to support reviews of topical reports.

So, again, we're trying to ensure that our tools and models are ready for licensing actions. So I'm going to talk more today about some of the work that we've done since 2019 to develop the necessary modeling capabilities in FAST to model LWRs, as well as to perform some assessments against the data that is out there.

And before I move on, I'll just, I want to make clear that I'm talking about thermal mechanical performance, so stress, strain, heat transfer, fission gas release, those types of things. So I'm focused on solid fuel forms, for molten salt fuels that's a little bit different, that's covered by what Steve was talking about earlier or what you'll hear next on the volume three source term analysis. So, again, talking about solid fuel forms here.

So, I apologize, this slide is pretty busy but this is taken from a presentation that was at the RIC just to highlight what the FAST fuel performance code is. So FAST itself is relatively new but it's built on FRAPCON and FRAPTRAN which are a lot older, going back for a few decades. So FRAPCON, FRAPTRAN, and now FAST were built for LWR fuel analysis, they've since been extended to look at non-LWR concepts but, yeah, a lot of the work that was done in the past is focused on LWRs.

But the codes have been extensively validated for the data we have for LWRs, and they're used quite extensively both domestically and internationally. So it provided a good starting place for us to move forward with non-LWR analysis. And I'll also say that these codes, or the FAST code is developed by Pacific Northwest National Lab primarily, but we do some of our own analysis and code development in-house at NRC.

Okay, so moving on to non-LWR fuels. Our prime goals have been to update FAST with the relevant models for metallic fuel, focusing especially on uranium, plutonium, zirconium metallic fuel alloys. Because that's what we have a lot of experience with in the past, and that seems to be the predominant alloy of interest moving forward, at least for the very near future. And then also looking at TRISO fuels.

And then, once we've gotten far enough along with some of our code development work, the important thing, of course, is to assess it against available experimental data. And, fortunately, there is a fair amount of data out there for both metallic fuel and for TRISO, certainly nowhere near the amount that we have for LWR fuels but still enough to help us assess our codes.

MEMBER PETTI: James, this is Dave.

MR. CORSON: Yeah?

MEMBER PETTI: Just a comment on the metallic fuel. There's an application in-house and it's no plutonium in it, so it'd be really good to make sure you've got data for the uranium zirconium alloy, that's where the earliest focus will be, I believe.

MR. CORSON: Yeah, you're correct about that. And I should have made that clear, this is, it's not only with plutonium, it -- in fact, I think a lot of the models are probably more applicable to just U-10 Zirc as opposed to UPU-10 Zirc. But we do have models that should be able to handle a range of plutonium fractions, going from zero to, I don't know, 20 percent or so. I forget exactly how high they went in EBR-II days.

MEMBER PETTI: And then, of course, the claddings are different, you know, you go back to these older alloys, what will be used today.

MR. CORSON: Yeah. So, I think I have this on the next slide but I'll just say it now, we focus primarily on HT-9 right now, because that seems to be what the most interest is right now. But, as you say, I mean, there's some tests that had D9 cladding going even further back, you know, SS-304, I think, or 316. I forget, but the more traditional stainless steel claddings. So, yeah, I think our models are primarily focused on HT-9 for now.

(Simultaneous speaking.)

MEMBER PETTI: Good.

MR. CORSON: Okay. Yeah. So, as I said,

you know, there's a fair amount of data from EBR-II and, to a lesser extent, Fast Flux Test Facility for metallic fuel. And then there's quite a bit of data from DOE's AGR program for TRISO that's been going on for about two decades now, maybe even a little bit longer dating back to the NGNP days.

Okay, so first I'll say, you know, when we wrote our plan in 2019 we had very basic capabilities for metallic fuels, extremely simple models for fission gas release and swelling, as well as some material properties like thermal conductivity, thermal expansion, and so on. Since then, we've done some evaluations to see what other models we need or what improvements we can make, and so far in the last few years what we've really focused on is improving our fuel swelling and fission gas release models.

So our models are still very empirical, moving forward we would like to do more mechanistic models. But, for now, the empirical models seem to work pretty well.

So, on the top-right, this just shows the curve fit for fission gas release for uranium, plutonium 10 Zirc fuel. So, the dots, this is a pretty, I guess, common graph showing results for a range of plutonium fractions. I think, in fact, some of these are just U-10 Zirc with no plutonium. So the, you know, the simple curve fit works pretty well. There is a little bit of uncertainty, certainly, but for now the empirical fit should work pretty well.

The anisotropic fuel swelling model, also empirical, a little bit harder to visualize in a few plots because it does account for plutonium fraction and, you know, burn up and so on. But yeah, that's something else that we've added to the code. I don't have it on this --

MEMBER PETTI: So James, just a question on the swelling.

MR. CORSON: Yes?

MEMBER PETTI: You know, all this data is on really shorter rods and I just, I don't know how it scales well to longer rods that will be in actual, you know, applications that are going to come in. But I did find a more recent publication that is a more sophisticated fuel swelling and fission gas release model, kind of together. And it supposedly does a lot better, it's a little more fundamental and not as empirical. So you guys might want to look at that, I believe it came from INL.

MR. CORSON: Yeah I, we very much, you know, pay attention to what is going on in the NEAMS program, or INL in general. And as much as possible we'd like to leverage what's out there, to put in our codes. I think for TRISO, as I'll say, that's an example of where we really have leveraged a lot of the work that's been done by DOE, INL in the past.

And yeah, I think ideally we would do the same thing moving forward. We don't have the same resources to develop these models ourselves that INL has, but as much as possible we'd like to learn from them and use their models when appropriate.

But I think, you know, you bring up a really good point and, about, you know, the limits of the existing database. We know a lot about fuel that looks like EBR-II, but what happens when you change things like sphere density or, as you said, the height of the fuel, active fuel length, operating temperatures, so on? We know a little bit from the historical evidence, but the uncertainties get quite a bit larger once you start deviating from our historical experience.

So that is why the more mechanistic models will be important, but I think we'll still need some sort of data to, hopefully, validate them.

MEMBER PETTI: The other thing is just, you know, beyond just the sort of steady state performance, do you envision using FAST for some of the transient performance, the overpower protected events and the like, to show, to confirm a fuel's going to be okay?

(Simultaneous speaking.)

MR. CORSON: Yeah --

MEMBER PETTI: Because that's a more sophisticated calculation.

MR. CORSON: Yeah. To some extent we would like to use FAST. I think, you know, for LWRs the way we do things, for the most part we use FAST for steady state type performance, and then occasionally we'll get into using it for LOCA or reactivity initiated accidents if we have some questions about the detailed fuel performance. But for the most part we can get away with using something like TRACE, a systems code, or, you know, code like MELCOR, to do those sort of transients.

So the answer is yes, we would like to develop the capabilities in FAST. But I think the more simplified approaches in the systems codes may be sufficient.

MEMBER PETTI: I worry that, you know, the FAST reactor transients, that's not, it's not going to work. You're going to need FAST, I think. You going to have to deal with the creep, you know, the pressure on the cladding relative to the expansion. That's why this model is so important, you know, the fuel pushes on the clad but it also extrudes up the clad. How much it does of each is a knob in the code, as far as I understand --

(Simultaneous speaking.)

MR. CORSON: Yes.

MEMBER PETTI: Except that there's this new model which gave me hope that there's something more phenomenological out there that could help think about how to scale it. Because to me that's, you know, we're not going to be able to do a transient test of a current fuel that the applicant is proposing, because where are you going to get the damage on the clad? It's going to all be, you know, lightwater reactor, you'll be lucky to get a couple DPA, that ain't interesting.

So, you know, the modeling is critical, it'll be critical for the applicant. And so I think it's going to be critical for the staff to have some confidence in those calculations, so.

MS. WEBBER: Maybe that's something, if you don't mind, you can send to us because --

(Simultaneous speaking.)

MEMBER PETTI: Yes, I was going to have these things --

MS. WEBBER: -- James may have it already, but we can just double check --

MEMBER PETTI: Yeah, I was going to send it to Hossein to put on our SharePoint, I can tell him to send it to you guys.

MS. WEBBER: Thank you.

MEMBER PETTI: I dug up some stuff that may be useful.

MR. CORSON: Yeah. I mean, that'll be helpful. I think, you know, usually if something is in Journal of Nuclear Materials I see it and flag it, but yeah, some things do slip my notice.

MEMBER PETTI: This one was in a weird one, I'd never heard of that journal --

(Simultaneous speaking.)

MR. CORSON: Yeah, that seemed -- yeah.

MEMBER PETTI: It was an odd one, so.

MR. CORSON: Yeah. So that -- yeah, if you have stuff like that, that would be really helpful that, you know, I haven't come across myself.

So, yeah, it, you know, we started to get into this a little bit, but we still need to do a little bit more work looking at the fuel failure models so that we can do more transient type analysis. We also need to add a fuel-clad chemical interaction model. Likewise, you know, we're probably going to at first do something pretty empirical, based on the type of data that is shown here on the bottom-right. But, again, you know, we would like to have more mechanistic models, and we do look to our colleagues at the labs to help out in that respect.

And, yeah, like I said, at the bottom, more mechanistic swelling and fission gas release models. So, certainly, if you can send us the information you have, we'll look it over and maybe that can inform our own models.

MEMBER MARTIN: This is Bob Martin. To your point about more mechanistic models, a code like BISON has been invented for that purpose. You know, the goal should not be to make FAST-BISON, I think the emphasis on, you know, how you use and implement empiricism based on new data, what have you, is extremely valuable for analysis because it's the best knowledge, maybe, at the time.

I wouldn't want to see you lose the ability to have those empirical models in there, at least as an option. You know, you might want to get, replace one with a mechanistic model at some point but code options, and I'm feeling my age, I think it's nice to be able to move back and forth. And at the same time, you don't want to, if you keep on going down the path and make it look like BISON, well then, you'll get rid of FAST and everyone will be on BISON. So you got to keep the personality of the tools, you know, unique, you know, because there are unique applications for FAST and --

(Simultaneous speaking.)

MEMBER PETTI: But for instance, you could paramaterize this new model, right, and could fit the whole darn thing and stick it into FAST. You know, I mean, and you can look at what's important, what's not important in there, it just gives you some insight as to whether or not what you have is good enough or you need to extend it.

MEMBER MARTIN: Yeah.

MEMBER PETTI: Yeah. No, I agree with you, you don't want this to become BISON.

MR. CORSON: Yeah, that's exactly right. And, in fact, you know, we're working right now on a slightly more detailed fuel swelling model that does have more parameters than what we have right now, and we are adding it as an option to the more standard model. So we're already doing exactly what you're suggesting.

And I agree, you know, we, from the start we never wanted to recreate BISON because we just don't have the resources for that for one thing. And another, you know, there is a place for a more simplified empirical analysis, we don't have the same responsibilities, I guess I would say, as the vendors do for their own analysis tools. So, yeah, I think as much as possible we have tried to keep things a little bit simpler, based in part on ACRS's feedback in past meetings. I think that's been really helpful in guiding our own efforts.

Okay. So, unfortunately, the assessments that I'm showing here are pretty dated, these are dating back to 2018. As I said, we had pretty simplified models at the time. But even then, with the very simplified models we do capture a lot of the behavior that's important from the, you know, especially the colliding strain, that's what we're pretty concerned about when it comes to fuel failures, and so on.

So, we're in the process right now of updating the past assessments that we've done. We only had, I think, four cases that we've looked at in the past but, you know, Argonne National Lab has a great database of the old EBR-II data. So we'd like to expand beyond our, you know, the four cases that we've done, we want to redo those and then expand to the, you know, several dozen cases that are available to us.

So there is still some more work to be done here. I think, you know, our assessments so far have shown that FAST does pretty well for the steady state analysis, but with these better models we're hoping to reduce uncertainties in our predictions.

So, moving on to TRISO. This is something that in 2019, when we presented our plan, we didn't have anything yet for TRISO fuel. So, we had talked about, you know, having TRISO models in FAST, ultimately perhaps we will end up incorporating that in the main version of the code but for now we just have a standalone code, a simple 1D code for TRISO fuel performance.

For those of you who are familiar with PARFUME, what we're doing with FAST TRISO is pretty similar to that. And we've leveraged a lot of the work that was done for PARFUME in terms of the various material properties and, you know, the solution for the mechanical stresses in the layers that were done as part of that program.

So, the last release of FAST TRISO was from a couple of years ago now. It was pretty simple, we could do heat transfer and fission product transport, some very basic stress calculations in the layers, it didn't account for the layer swelling and creep, that's really important. So it wasn't in that version of the code but it did have some Monte Carlo analysis capabilities to calculate failure probabilities. So there is, you know, some work that's needed to be done from the last version of the code that was released.

Now, fortunately, quite recently, in fact, we did implement the mechanical model used for pyrolytic carbon swelling and creep. And so at the bottom, this is just showing comparisons to this IAEA coordinated research project CRP6, it had some simplified TRISO fuel cases and asked the participants in the benchmark to do these simplified calculations. So you can see below, you know, now our calculations for layer stresses are pretty close to what BISON is getting for these idealized cases.

So, the one outstanding development item is to develop the stress correlations that allow you to capture multi-dimensional effects in the simple 1D calculation. So, this counts for things like the pyrolytic carbon layer cracking and de-bonding, and the stresses that that would impose on the silicon carbide, and can also account for spherical particles. So we're -- this is ongoing right now, we're hoping to have it done in the next couple months, to incorporate in the, in our version of FAST TRISO.

And then, once we do that, of course, we need to expand our assessments. We've done some very simplified calculations of fission product releases from AGR, I think from the AGR-2 set of tests. So, we need to repeat them once we have the more, once we have the improvements made to the model.

So, this last slide just sums up the work that we've done in the last few years. So, our codes are ready to do confirmatory analysis for metallic fuel and uranium oxycarbide TRISO. Obviously there's more development work that would help reduce uncertainties, and we, of course, need to do more assessments to gain confidence in our models, but nevertheless we do have capabilities to do some confirmatory analysis.

That doesn't mean that we're done, we would like to add more mechanistic models, as I've said. Again, it's not going to be recreating BISON, but we could take into account more parameters, perhaps, that do influence the thermal mechanical performance.

But I'd say, in closing, one of the most important things of this activity is it has really helped build staff expertise in this area. It's one thing to take a model off the shelf and use it, it's another thing to be involved in creating that model and understanding what goes into it, and all the limitations. So that exercise, I think, has perhaps been even more valuable than the code development efforts itself. It's really helped us understand what's important and we'll be able to use that when we're supporting licensing actions that NRR has to take.

So, that's all I had for my presentation, and I'd be happy to take any questions.

MEMBER MARTIN: One -- of course, I see the statement about EBR-II and AGR. Are there fuel data sets that are out there that you should be gathering in and incorporating into your, you know, co-development efforts, your assessment efforts, that, you know, just haven't risen in the level of consciousness yet and that should? I'm, you know, looking at more -- Dave maybe has what you've been --

(Simultaneous speaking.)

MEMBER PETTI: No, the only -- there are some FFTF metallic fuel --

MR. CORSON: Yeah, I --

MEMBER PETTI: I'm sure it's part of that database --

MR. CORSON: Yes, it is. I think --

MEMBER PETTI: It's dominated heavily by EBR-II, but there was some, so there's a length effect, because those are longer, so that's beneficial.

The only thing I had a question on, you know, what I found in the days when we were doing the TRISO modeling, you know, these rods are different, what are being proposed by the applicant. Their diameters are different, thicknesses of cladding are different, how much of an effect does that have? You probably have the capability to take, okay, here's EBR-II, here's FFTF, here's what the applicant's saying, you know, what's the, translate those physical dimensions into things that matter.

Like, what you think the clad strain is, you know, are they pushing the envelope or is there more margin? That stuff doesn't come through and that would be useful to NRR, I would think. And I don't think it's a difficult, those are difficult calculations to do, to run through those. Like, uh --

(Simultaneous speaking.)

MS. WEBBER: It's like a sensitivity study.

MEMBER PETTI: Yeah, sensitivity studies, basically --

MS. WEBBER: Yeah.

MEMBER PETTI: To see what's going on.

MR. CORSON: Yeah, I think that would be really beneficial to do, I agree. You know, so far we've focused more on our development efforts and to a lesser extent on the assessment efforts. But going forward, I think it would be useful to do those sorts of sensitivity calculations, start exercising the models a little bit more than we've done so far.

And I'd also say as far as like other assessments, so this pretty much captures the historical data, EBR-II and FFTF for metal fuel, AGR for TRISO. But there are some very active programs at DOE to generate more data.

So the advanced fuels campaign has done a lot of work in recent years on metallic fuel. And they continue to do tests. We participate in advanced fuel campaign meetings at NRC, so we're aware of what's going on.

And for both metallic fuel and TRISO, there was recently a proposal, a project proposal under FIDES, the NEA joint project Framework for Irradiation Experiments, to do some TRISO and metallic fuel irradiations at ATR. So of course, you know, it takes time to accumulate burnup, it's going to take some time before we get those results.

But NRC is participating in that project, so we will start to get more data on metallic fuel and AGR that differs a little bit perhaps from the historical irradiation database.

MEMBER MARTIN: One question for Kim. How formal has, you know, your division been in the maintenance of data sets? Is it something that, you know, every code team kind of has in your back pocket on a share drive somewhere? Or you know, once upon a time there was a database of sorts, and that got kind of loose support I think over the years, you know, from a maintenance standpoint.

What's the status of data integrity of the agency?

MS. WEBBER: I think it's a great question. I do think that at this time, that for this work, the data resides with the leads who are working on these codes. But internal to the division itself, we do actually have a data management strategy that we've just started to implement to try to collect data sets, put them in a centralized location, and try to, you know, maintain it to the best we can.

What we do realize is that most of the data is not ours. It's other organization's data. So when it comes to maintaining data, you know, there's the, it's kind of a slippery slope on what our responsibility is versus others' responsibilities.

So right now, you know, I would say that, you know, to the best that we can, we have databases of data, but that's representative of other people's data. Like James talked DOE's data and the national labs produce data, international data.

MEMBER MARTIN: Well, I think back in the day with light water reactor technology, you know, you had the data, you know, I don't know if it was database, it kind of went away. But it was tied to a lot of agreements, you know. And there were --

MS. WEBBER: Yes, it was.

MEMBER MARTIN: International agreements, what have you. And expect those to expire, which creates its own legal challenges, logistic challenges. It sounds like we still really haven't solved the maintenance question with data. And probably still having folks going to like old papers and stuff and digitizing.

MS. WEBBER: Yeah.

MEMBER MARTIN: Seems quite arcane.

MS. WEBBER: But the one thing that I have to say is that, and maybe others can speak to this, is for each of the major codes, like TRACE has its own manual that documents what data sets it's used to maintain its, you know, status of making sure the code runs with new features and so forth.

So that is a plan, you know, that we have is to document that. And you can see it in Steve's, you know, efforts, he's trying to document V&V, and that's a way to keep track of what data is being used.

The challenge that we have is so far our funding has been so focused on developing the data and acquiring data through these ad hoc methods to validate the codes themselves that we don't have funding, you know, to be able to do the, I'll call it fancier things that create our own database and make that accessible.

So I don't know if others at the table want to chime in on that, but.

MR. BAJOREK: This is Steve Bajorek. We're nowhere near the capability that we have for the light water reactor where we have and maintained our own database. The non-LWR data is more of an ad hoc basis.

It's with the code developers right now. We get snippets of it now and then. But we could not go to a central repository as we could for the light water reactor.

Argonne National Labs has been putting together one for that does include the EBR-II data and some other databases, that that looks like to be a good start. And there are some international efforts to start pulling together a non-LWR database, but they're still in their infancy right now.

But you know, as we go on, it is going to be important to collect that data, put it in a location that we can use it and keep it expert-controlled, proprietary, as it needs to be.

MEMBER MARTIN: Well, obviously it creates a challenge, not just obviously for the agency, but for anybody that's advancing in technology. I mean, certainly they would have to make those agreements, you know, to get access to the data. But the data has to be in a convenient spot where they can make an agreement and make a deal, bring it in, so.

MEMBER PETTI: I know there are databases under the Gen IV. And so you probably can get access through DOE. So and they break up by area. There's a whole big code area that's all about V&V. I know fuels, there was data that was sent many, many years ago. Labor, constitute relations, that sort of stuff. And I think it was done on all the systems.

MR. BAJOREK: Yeah, we do try to get involved in some of the international benchmarks, and that's often a good way of --

MEMBER PETTI: Yes.

MR. BAJOREK: Getting the data. We're getting involved in one from HTTF, there are some other ones that we're involved with. But that's a really good way of getting data without having to pay a lot of money for it.

MEMBER PETTI: Right, yeah.

MEMBER BIER: I have a quick question for James. This is Vicki Bier. For one or two of the fuel types where you said you did not yet have fully mechanistic or phenomenological models, but you were doing Monte Carlo simulation, can you talk about what that is actually simulating? Is it just empirical, or how is that organized?

MR. CAMPBELL: Yes, so for TRISO fuel, this is actually something that's done in part because of the nature of ceramic material behavior. So you have to do a statistical analysis to calculate what the fuel failure probability would be.

And so it can sample on anything from the layer thicknesses, which that comes from manufacturing, it's usually known what the variability might be in the layer thicknesses, to some of the material properties we can also sample.

Those are maybe less well-defined what the distributions would be. But there is some information about how some of the material properties vary a little bit.

But yeah, that's what we're using the Monte Carlo analysis for, to calculate the probability of the pyrolytic carbon and silicon carbide layers in TRISO. And this is a capability that, you know, PARFUME has. BISON can do this as well. So it's a pretty common way to analyze TRISO.

MEMBER BIER: Okay, thank you very much.

MR. CAMPBELL: Sure.

MEMBER MARTIN: Any last questions on this subject before we move on to the next?

MS. WEBBER: Okay, so thank you, James. Have a safe travels.

MR. CORSON: Thank you.

MS. WEBBER: And next I'd like to introduce Shawn Campbell. Shawn Campbell's a Reactor Systems Engineer in Hossein's branch, again. And Shawn, and Lucas Kyriazidis is here to support as well as Andy Bielen. So Shawn will be the main presenter, and then Lucas and Andy will be able to answer questions if Shawn's not able to.

So take it away, Shawn.

MR. CAMPBELL: All right, we'll do a quick mic check first. Can everybody hear me okay?

MS. WEBBER: Yep, you're great.

MR. CAMPBELL: Okay, great. And just a quick check on the slides as well. Can you see those all right?

MS. WEBBER: Yep.

MR. CAMPBELL: Okay, fantastic, thank you very much.

All right, well, good morning, everyone, and thank you for giving us this opportunity to share with you some of the work that we've been doing on our codes to prepares our codes for a non-light water reactor application.

So as Kim said, my name is Shawn Campbell. I'm joined this morning by my colleagues Lucas and Andy. The three of us work in the Fuel and Source

Term Code Development Branch in the Office of Research. And our branch is primarily focused on the SCALE, MELCOR, and FAST codes.

And you've just heard from James on the FAST code. And today we're going to talk about, for this next presentation, we're going to talk about the SCALE and MELCOR codes.

Before I get started this morning, I just wanted to take this opportunity to quickly recognize our colleagues at Sandia National Labs and Oak Ridge National Lab. Our partnership with our SCALE and MELCOR code developers at these labs has been instrumental in the success of this work. And so I just want to say thank you to them and give them recognition for the work that they've done.

And then also just to let you know, we do have several of the code developers online with -- on this call. If there's any specific questions associated with the models or anything, just so that you know that they're available for that. And I'll go to the next slide. Sorry.

So in our approach to Volume 3, we had a few key objectives in mind. First, we really wanted to better understand the severe accident behavior of these various non-light water reactor designs.

And with that better understanding provides some insights to the NRC's development of regulatory guidance. We wanted to build the knowledge and the expertise among the NRC staff on the modeling capabilities that we have for these non-light water reactors.

Our next objective was to encourage dialog among the various stakeholders on our approach to applying SCALE and MELCOR for source term analysis and get early feedback. And we did this by hosting public workshops for various reactor designs.

Our third objective was to ensure that our codes are ready. That's been a big topic today obviously. Ready to support non-light water reactor licensing. And so for this, to do this, we have developed modeling capabilities in SCALE and MELCOR, and we are able to identify accident characteristics and uncertainties that may affect the source term.

We also developed publicly available input models for each class of non-light water reactor that we can make available upon request.

While the Volume 3 report and overall approach was developed in the 2019/2020 timeframe, I just wanted to point out that we've been working on developing our SCALE and MELCOR computer codes for non-light water reactor applications for quite a while now. So for example, back in the NGNP days, you know, 2006-2013, we outfitted our codes with a lot of capabilities for TRISO fuel and HTGRs at that time. Next slide.

So I'm sure you're aware that I wanted to give you a very high level understanding of the codes that we are using here. This is a slide that we showed at the recent RIC. It was a poster, a digital poster that we had.

SCALE is the NRC's comprehensive neutronics package. It's developed, like I said, by our contractors at Oak Ridge National Lab. Some of the key capabilities of this code are nuclear and data cross-section processing, decay heat analysis, criticality safety, radiation shielding, radionuclide inventory, depletion generation, reactor core physics, and so on.

You can see here SCALE has a very wide user base. It's used not just by the NRC but used by 61 countries around the world, with 11,000 users worldwide. So, very wide user base. So it's been exercised quite a bit.

It's also a highly validated code. It's been validated against numerous shielding depletion criticality, etc., assessments. And so it has a strong pedigree associated with it.

MELCOR is the NRC's severe accident progression and source term code. This one's developed by our contractors at Sandia National Lab.

This code's able to simulate the accident progression and thermal response of the reactor, the model of the reactor heatup, the degradation and relocation of the core as it degrades. Track the release of the fission products from the fuel, their transport through the reactor as it goes through the vessel to the containment and then out into the environment.

Like SCALE, MELCOR is used domestically at universities and laboratories and so on. But it's also distributed throughout the world. We have over 30 organizations internationally that are using the code. And it's distributed through our cooperative severe accident research program.

MELCOR also has an extensive validation associated with it. It's been validated against numerous international standard problems, benchmarks tests, and integral experiments over the years dating back from the 80s to, all the way to today.

So shown here was our overall project

approach. So our approach, like I said before, was to develop workshops for various reactor designs. So our first step was to build representative input models.

So using what publicly available information we could find, we had Oak Ridge National Laboratory build detailed core input models in SCALE. And then our counterparts at Sandia National Laboratories built full plant input models in MELCOR.

We then proceeded to select plant accidents that we thought would best demonstrate the capabilities of our new models that we implemented in these codes.

And finally, we performed a series of simulations with scale modeling, things like decay heat, radionuclide inventories, reactor BT, back coefficients, and so on. And then feeding those as inputs into MELCOR and then performing full accident progression and source term analyses.

And then as we -- as appropriate, we did quite a few sensitivity analyses as well for these various designs.

So shown here is our overall project scope. We had five major non-light water reactor types that we investigated. For each of these, we held a public workshop to describe the unique features, describe the new models that we had implemented, and provided the results for our analyses and sensitivity analyses.

On the left here for each one, we give the reactor type, and then on the right we show the specific design that we used in our analysis for the demonstration project. The reference reactor was chosen really based upon the degree to which we could find publicly available information.

And in those situations where we didn't have specific information, for example, design of the containment and leak rate and so on, we just, we used our best judgment in creating those.

So back in 2021, we held three workshops. The first one we did was for a heat pipe reactor. And for this one we did the INL design, the concept reactor.

For the high temperature gas-cooled reactor, we used the pebble bed PBMR 400. And then the last one we did in 2021 was a molten salt cooled but still pebble bed geometry. For this one, we did the UCB Mark 1.

Moving into 2022, we conducted a workshop for a molten salt reactor. This one's a molten salt fueled reactor. So this is the MSRE design. I think

Steve talked about this design before. And then also we did the sodium fast reactors. This one was the ABTR.

All of these workshop materials can be found on our public web page. We have a couple ways to get there. You can click on this link if you have the slides. Or scan this QR code. And this is a snapshot of what it will take you to.

We have all the slides put up for these workshops. We have YouTube video recordings. And then we have SCALE and MELCOR reports, and these reports go into extensive detail on the design, the reactor designs, the models that we created and the analyses that we conducted, as well as sensitivity analyses. So those reports go into a lot more detail than you'll even find in the workshops.

So from here, I'm going to provide a high level overview of the content of these five workshops. Like I said, if you want more details, I encourage you to go to this webpage and explore some of this. And at any time you're welcome to ask any questions about what you find there.

So the first one I wanted to go into more detail on is the fluoride salt high temperature reactor, or the FHR. So this one was a 236 megawatt reactor. It uses Flibe for the coolant and has a TRISO fuel pebbles and a pebble bed geometry.

The pebbles are 19.9 weight percent. It undergoes online refueling and operates at atmospheric pressures. I'll just point out the direct reactor auxiliary cooling system, or DRACS, is made up of three trains of passive heat removal systems, each with a capacity of about 2.36 megawatts, or around 1% of the full plant power.

Each train has four natural circulation loops, as you can see over here. The first train goes here. There's a ball valve that drops whenever you have -- the differential pressure falls whenever you have a pump. And the coolant, the primary coolant is diverted into this first heat exchanger, which is also a molten salt.

And then this one is the -- your first loop goes over here into a water loop, and then finally into an air loop or a -- which is just a stack. All of these are buoyancy-driven flow, there's no pumps. And so it's a completely passive decay heat removal system.

Shown here are the three accidents that we modeled for this workshop. We did an anticipated transient without scram. So for this one, it was a loss of onsite power and then a failure to scram.

So, all of the pumps tripped, reactor failed to scram. Secondary heat removal ends, and then we have anywhere from zero to three trains in the DRACS operating, so we investigated the ability of DRACS to remove the heat.

The next accident was a station blackout, which is kind of self-explanatory. But complete loss of power. Salt pumps trip. And then your heat removal ends and variable amounts of DRACS. And one again to see how this scenario plays out.

And then our final scenario was a LOCA. And so for this one, there is a three-inch line up here. We don't have it pictured. But there's a drain tank up here on this line. And so there's this three-inch pipe that comes off into the drain tank. And so we assume a break of that line.

So we varied the size of that break up to the full break, full pipe of three inches. So for this one again, we looked at variable DRACS and looked at the response of the plant.

So shown here are some of the new features for SCALE and MELCOR that we added to the codes to facilitate this demonstration project. Over here in SCALE on the left, we incorporated that new interface for more efficient depletion calculation for TRISO fuel. And so this made it easier for us to perform sensitivity analyses.

We also leveraged a workflow that we had developed for the HTGR demonstration project for modeling TRISO in what we call SCALE/TRITON.

On the right, we added a generic framework for inputting working fluid equations of state. We added fission product chemistry transport models for molten salts. Improved on the fission product release models for TRISO that we had originally developed for HTGRs. And then added point kinetics enhancements for reactivity insertion transients.

Shown here at the bottom are our cutaways of our SCALE and MELCOR models. You can see over here -- I'm always impressed by the scale graphics that they're able to create. But here's the reactor core model in scale and a slice of the -- coming from this model. And then you can see here one of the TRISO pebbles with the TRISO particles on the outside and the graphite core in the middle.

On the right here you can see our MELCOR nodalization, with the core nodalized here, and then here's the, excuse me, the primary and secondary pumps.

MEMBER MARTIN: Quick question on capability.

MR. CAMPBELL: Yeah.

MEMBER MARTIN: Does MELCOR have multi-dekinetics if you needed that for a problem like this?

MR. CAMPBELL: I'm sorry, say that again, I couldn't hear you.

MEMBER MARTIN: Does MELCOR have like multi-dekinetic capability? I think you mentioned that it deployed kinetics as a good improvement. It just makes me ask the question if you needed more, is there more.

MR. CAMPBELL: It's still an ongoing. We do have a lot of capabilities. But as of now, our plant kinetics models are pretty basic. We have recently added capabilities also for, you know, dissolved fuel, right. So you have your delayed neutron precursors, and be able to track all of those as well.

MEMBER MARTIN: I was specifically asking just about kinetics. Is there a 1D, 2D or whatever it is?

MR. CAMPBELL: Right now it's all 1D.

MEMBER MARTIN: Okay. Or zero-D.

MR. CAMPBELL: Oh yeah, zero-D, sorry,

zero-D, yes.

MEMBER MARTIN: Okay, thanks.

MR. CAMPBELL: All right, I'll move on to the next slide here. So on this slide, where I provide some of the typical results that we received. These are high level insights that we obtained in these scenarios.

As I mentioned before, for ATWAS, the fuel heatup was limited by reactivity feedback. So this is primarily the fuel temperature feedback that prevented the -- too much fuel heatup. The passive decay heat removal system DRACS was also effective in removing heat, as you can see here.

With even a single train of DRACS available, we were able to remove the decay heat and prevent fuel heatup. It's only when we have all three trains unavailable that we see any real fuel heatup.

For SBO, we had, if there was complete failure of the DRACS, then we did see the coolant boiling occur. But it was really over the course of several days. As you can see, this is a very slow moving transient over here.

And then for LOCA, again, a single train of DRACS was sufficient to prevent any fuel damage. And only when all decay heat removal was unavailable did we see any coolant boiling followed by fuel damage.

Over here in the case with no DRACS available, we did see some release of cesium. This is the release rates that we see back there. Cesium release from the pebbles to the liquid molten salt starts earlier over here because of the heat at lower temperatures. You can see it's a very small amount until we actually get any real fuel heatup.

All right, so back in 2021, Kairos submitted a construction permit application for their Hermes 35 megawatt nonpower reactor. So at that time, we were approached by NRR to perform some scoping calculations to explore DVA level transients. And so by that I mean we're not really exploring core damage or fission product release transients in the -- here.

So the MELCOR FHR reference plant model that we had -- that I just discussed was modified to support a quick turnaround set of calculations to support the review of the construction permit application for Hermes.

These analyses provided insights on the relative importance of potential accident scenarios and focused the license review on the most safety-significant topics. The two base scenarios that we looked at here were a loss of force circulation.

So this is a concurrent trip of the primary and intermediate coolant loops. And then we also looked at an insertion of excess reactivity. And this, for this one it was an accidental control rod withdrawal. I'll just point out that we have presented this previously during the Hermes construction permit ACRS meeting.

So on the neutronics side of things, we used SCALE KENO for the multi-group Monte Carlo transport and origin for the isotopics. We did use a random pebble geometry, and we approximated that by a regular lattice. Equilibrium isotopics were generated iteratively through a two-dimensional slice models in our SCALE/TRITON code.

And over here on the right you'll see that we really got excellent agreement between our results and Kairos', given the information that we were able to glean from the PSAR. So we were pretty pleased with these results.

And then on the MELCOR side of things, like I said before, we used the UCB Mark 1 MELCOR model as our starting point and then adapted it to be a little more Hermes-like. We focused our efforts on the primary system.

And the secondary system and the decay heat removal system here were really mostly modeled by boundary conditions, just because of the lack of detail we could find in the PSAR.

But the DHRS model uses a -- uses water at a constant temperature with a boiling heat transfer coefficient here for the evaporator tube wall. To the right you can see the schematics that we have from the PSAR, and we used these to develop our models.

I'll just note here the DHRS is -- well, it's a different design, of course. It's analogous to the DRACS system that we saw before in the UCB Mark 1.

So here I just wanted to show some of the results from our two base calculations that we did in doing the Hermes scoping analysis. On the left here is the insertion of excess reactivity transient.

So for this one, there's a rod withdrawal, it's the highest worth rod that we assume is withdrawn. So we get about three dollars' worth of reactivity inserted over 100 seconds.

So here the reactor trips on high power. That's about 120% power. And that occurs at about nine seconds. And concurrent with a PSP trip.

As you can see, the temperatures here all remained within the safety envelope proposed by

Hermes. And you can see up here is a snapshot from their PSAR. And you can see we got pretty comparable results to Kairos.

Also same thing over here on the left -- on the right, sorry. We have a -- the loss of force circulation scenario with a concurrent trip of the primary intermittent coolant loops. And again, all of the temperatures remained within the safety envelope, and our results are very similar to those that were predicted by Kairos.

MEMBER MARTIN: A question I can't help but ask, how do you model pebbles with MELCOR?

MR. CAMPBELL: How do we model individual pebbles, or?

MEMBER MARTIN: Well, I mean how do you model the core, and then you could break it down from there.

MR. CAMPBELL: Okay, complicated question.

MEMBER MARTIN: You don't a have a coarse/medium type solution. So you have traditional finite volume type modeling, correct, and you're using, you know, simple geometry, each structures. But yeah, you do report out like max TRISO.

Is that -- that's the truly at the kernel level type solution? So there's some fidelity down to a very local level?

MR. CAMPBELL: I think that like I said, this is a as you know, MELCOR is a long parameter code here, right. And so we're getting a lot of this and we're having to kind of smooth it over these individual volumes. Let me show up here.

So for each of these we have individual core nodalizations, right. And so for each of these, we're getting a lot of the power density and so on, we're getting a lot of that information from SCALE.

So we're really reliant on SCALE to get a lot of that information and feed that directly into our MELCOR models in this lumped application, if that makes sense.

MR. WAGNER: Shawn, maybe I could jump in kind of quickly here.

MR. CAMPBELL: Yeah, sure, go ahead, Casey.

MR. WAGNER: So we have sort of a lump model for the bulk core behavior. We model the balls we have you know sort of a porosity solution for the pressure jobs that's Reynold's and porosity-based.

But for the peak fuel temperature, we used a -- we modeled a single pebble in the hottest spot in the core to, you know, high fidelity. And so all the layers, the heating on the inside at the maximum heating rate. And then we used that as boundary conditions for an individual TRISO that would have been at the inside of the annular region of the fuel.

And so the TRISO codings are all modeled individually in detail in the heat structure, with a boundary condition from that individual pebble that's in -- modeled with the heat structure. And so in that way, we were trying to get a lot of detail and a good prediction of the peak fuel temperature.

And so it is actually the kernel. We also have the individual layer temperatures too.

MEMBER MARTIN: Okay. Now, is that -- did that require development in, you know, whatever, the last ten years? Or was that the capability that's always been there with MELCOR?

MR. WAGNER: That capability's always been there. We don't typically add in a heat structure into the core package. And it's sort of a -- it's not relevant from a thermal hydraulics perspective, but it's very relevant from a monitoring peak temperature. You know, because it's only one wall.

And so I can put one ball anywhere I want or you know, across the core, and be able to model all the way down to an individual TRISO in the layers.

And so that heat structure capability has been there since the beginning of MELCOR.

MEMBER MARTIN: Yeah, I think of course maybe a different design or maybe this design under a certain situation where radiation's important. How do you capture view factors and all that? Is that readily accessible from the user standpoint to get that in there?

MR. WAGNER: Yeah, yeah. So from the heat structure model it has radiation and convection models, you know, for the outside surface of the pebble. In this case it was, you know, were covered in fluid, so that wasn't too relevant.

But I actually leveraged the heat transfer coefficient that, you know, the basic core components are modeling to patch that in as boundary conditions for the pebble, with passes boundary conditions for an individual TRISO.

MEMBER MARTIN: But from a standpoint of radiation, is there, I mean, is there a modeling that saves the user from having to figure out all the view factors?

MR. WAGNER: Nope, we have to put in view factors and consideration of the radiation. But there's a couple different types of models there. We can do -- I mean a radiosity model, which you know, if we had some information on that.

So yes, we approximate that.

MR. ESMAILI: Can I jump in? Sorry. If your question is about how we do model the pebbles versus the cylindrical fuel rods, the capability has always been there. It's fundamentally no different in how we are doing that, you know, straight fuel rods. The radiation is there, conduction is there.

Then you got to these pebbles, you know, it's like Casey and Shawn were saying, that then we have to model it a little differently. You know, like we use like for example Ergun equation to calculate the you know, pressure dropped points through this.

Fundamentally it's very, very similar to what we are doing. We didn't fundamentally change how we are doing things in the core package.

As a matter of fact, many years ago James Corson used the existing capability of MELCOR to do a HTGR and now he, you know, during the NGNP times and we built on that. So think that capabilities were there if that's what --

MEMBER MARTIN: Yeah, I was really just, you know, wondering. I just think with pebbles it's a lot harder to get that right, radiation right. I mean, when you're dealing in prismatic, you know, geometries are still pretty simple.

And it'd be nice, I guess, if there was a, you know, some convenience incorporated into the modeling capability to make sure that's done right. You know, making certain assumptions about the arrangements of the pebbles and you know, the packing, what have you.

But it can certainly be done outside of the code and incorporated in the input that you described. It's just work, that's all.

MR. BAJOREK: I think for this situation, radiation probably should not play a --

MEMBER MARTIN: Right, that's why I mentioned a different, yeah, a different design might have that.

MR. BAJOREK: But when you have a pebble bed, you're getting a sort of a conjugate heat transfer. Could be by radiation gas-cooled react by convection, also conduction through the pebbles.

So what you should be using is like I think it's a Zener/Schrödinger type of model that accounts for all of that stuff as Jose pointed out, like a KTA or Ergun equation to get the pressure drops correct.

MR. ESMAILI: Those models are already in the core. We did that 12, 13 years ago as well.

MEMBER MARTIN: Thanks.

MR. BEENY: Hi, this is Brad Beeny from Sandia Labs. Yeah, I just I wanted to remind everybody yeah, we -- I think somebody just said it. But we do use the Zener/Schrödinger/Bauer with the Breitbag Barthes radiation term to account for the effective conductivity when computing heat transfer from within the core.

So if that's what the question is, how do we account for heat transfer within the core, that's what MELCOR is leaning on primarily with its core components, is this effective conductivity model that accounts for, as it was said, radiation, convection, conduction. This -- that unit cell concept that's -- yeah, that's in the code.

And then likewise, the Tanaka Josaka model for the prismatic version, if there were any questions about the other kind of HTGR.

MEMBER MARTIN: Yeah, thanks for the clarification.

MR. CAMPBELL: Thanks a lot, Sandia, I appreciate you guys jumping in there for that question. Is it okay to move on to the next one?

MEMBER ROBERTS: I have a question on the bottom left. It says that its first three odds in reactivity in 100 seconds, but it trips at nine seconds?

MR. CAMPBELL: Correct.

MEMBER ROBERTS: So it's a total reactivity insertion 9/100th for three dollars?

MR. CAMPBELL: The total was three dollars. It was done in a linear rate over 100 seconds. We reached the trip at 100 seconds. Or sorry, we reached that trip at nine seconds.

MEMBER ROBERTS: Right, so most of reactivity insertion occurred after the scram?

MR. CAMPBELL: Yes.

MEMBER ROBERTS: Okay. Did you look at a case with no scram? Where the three dollars actually got inserted?

MR. CAMPBELL: I'm trying to remember. I think that was one of the sensitivities that we did look at. I don't -- no, actually, I don't think we did. I don't think we did.

MEMBER ROBERTS: Okay, so next question is why? As is because it was not a design basis --

MR. CAMPBELL: Correct, yeah.

MEMBER ROBERTS: And you were limited to

design-basis events in this comparison?

MR. CAMPBELL: That's right.

MEMBER ROBERTS: Okay, thank you.

MR. CAMPBELL: Yeah. Yeah, we ran several sensitivities calculations. That's why I hesitated in responding. But no, because we were trying to stay within the confines of design-basis, we stuck with this, so.

All right. So moving on, in September of last year, the NRC staff accepted the Hermes 2 CP application. So we are currently, this is ongoing, we're currently supporting NRR's review of the application for Hermes 2 by modifying the Hermes 1 model.

So again, we're performing DVA level scoping calculations here. So I won't go into too many details here just because this work is ongoing.

MEMBER PETTI: So Shawn, just as you do that, think about whether there's a different event because the loop. You know, it's not just repeat all the ones from Hermes 1 again. But does the presence of that secondary system cause a new event to occur that you could potentially analyze.

MR. CAMPBELL: Right, right. Absolutely. And then that's feedback for NRR as well during this review.

MEMBER PETTI: Yeah, that's the question I'm going to ask.

MR. CAMPBELL: Sure. And this is where we are building out some additional capability or additional detail on the secondary side for in this Hermes 2. So now that we have some more -- with the -- we're able to peek under the table a bit more for the Hermes 2 and get proprietary information. We have been building out this secondary side.

MR. BIELEN: This is Andy Bielen. I just want to like temper expectations, though, because given we were able to incorporate some more like detailed information from Kairos.

However, as you guys saw with Hermes 1, much of the detailed design work has been, you know, pushed off to the operating license stage. And we're finding that, you know, it's fairly similar approach for Hermes 2.

So we have some more information. It's not a revelation, you know, in the additional modeling detail this will have available.

MEMBER PETTI: So you have some physical properties for the secondary salt?

MR. BIELEN: No comment.

MR. CAMPBELL: Great, well, in the time that we have remaining, which is not a lot, but I did want to share with you at a high level some of the -- some details on the remaining four demonstration workshops that we did. So I'll try to go kind of quickly through these slides, but more information can be found on the website, like I said before on slide 7.

So just as we did for going from UCB Mark 1 to Hermes, we've -- we're trying to create these models so that we're -- we can readily adapt these reactor models to future applications for new reactor technologies.

So the next workshop I wanted to talk about was our high temperature gas-cooled reactor. The representative plant that we looked at was the PBMR-400.

So this was a 400 megawatt thermal design with a graphite moderated heated and cooled TRISO fuel. The model is based upon the OECD NEA neutronics benchmark project.

So because some of the new key modeling for SCALE was a new interface for rapid depletion of TRISO fuel for more efficient computational costs. This is the same approach that we -- I talked about back for UCB March 1.

For MELCOR, we have improved models for TRISO fuel thermal response, radionuclide diffusion, failure models, and -- and it's important to note that a lot of this is leveraged from the effort that we did back in the NGNP days.

MEMBER PETTI: Yeah, just Shawn, I can't let this -- those source terms are ridiculously high. There will be no gas reactor vendor ever come in and say that there's an accident that releases a tenth of a percent of cesium out of the core. It's off by at least a factor of 50.

I don't think it's your diffusion models. I'm assuming it's the failure rate that you assumed. This is -- predates probably the EPRI topical report that has the data that shows under these temperatures what sort of failure rates you can expect.

So just I want to be on the record that those numbers -- actually, I remember reading the report and looking at that and saying there's something that doesn't make sense, so.

MR. CAMPBELL: Sure, and I think we've tried to say it before, you know, we're not trying to necessarily say these are the exact accidents that are going to occur. These are not the consequences associated with these designs or anything like that. We're not trying to make those types of assertions. We're trying to demonstrate our co-capability.

And so but take all of these values with a grain of salt I guess is what I'm trying to say.

So some of our insights that we gained from this. We looked at -- we found that graphite oxidation from air ingress didn't have a -- didn't generate enough heat to really impact the fuel in this case.

We also found that decay heat dissipated pretty readily into the reactor cavity. And it was enough to limit fission product release from fuel failure.

If you look on the right here, we did some sensitivity cases to determine what parameters had the greatest impact on fuel temperature. And you'll see that the low graphite conductivity had the largest impact on peak TRISO fuel temperature for this scenario.

MEMBER PETTI: Shawn, can you guys handle steam ingress in MELCOR yet?

MR. CAMPBELL: Yes, when we did. We did do air ingress in this case. It depends on what you're talking about. If you're talking about in a gas reactor, yes, we have that capability.

MEMBER PETTI: So a steam generator tube leak, right.

MR. CAMPBELL: We could model that, yes.

Moving on to -- oh, I'm up two slides, I'm sorry. So next is the heat pipe reactor. This was the INL design A. It's a 5 megawatt thermal reactor. It has only a five-year operating lifetime. Over 1100 heat pipes cooled are fueled with a metallic uranium at 19.75 weight percent.

What's unique about this design is it has these control drums on the outside that rotate around the periphery of the core to change the neutron flux. Some of the new modeling capabilities that we incorporated for SCALE, a new multi-group fast spectrum library was included. And also new 3D visualization improvements.

For MELCOR, we added new thermal physical properties for sodium and potassium. We added heat pipe reactor specific models such as -- well, adding the working fluid heat pipe connection to the secondary heat exchanger, heat pipe failure models, and so on.

The transients that we looked at here for the heat pipe reactor included a transient over power, loss of heat sink, and unanticipated transient without scram. If you look at the workshop, it's only the transient over power that we included in the workshop, and the other two are described in the reports.

So like I said, the figures on the right here then are the transient over power scenarios. And some of our key observations here that were -- that after scram, heat dissipation in this reactor cavity really ended the releases from the fuel.

Heat pipe pressurization on failure really drove the release from the reactor vessel into the reactor cavity or the reactor building. And the reactor building bypass actually required two failures of a heat pipe.

So you needed one failure in the condenser region and another in the evaporator region to get a release of any fission products.

MEMBER PETTI: Did you model, I don't remember in this design, the liquid metal in the heat pipe running?

MR. CAMPBELL: Yes, yes.

MEMBER PETTI: Interesting, okay. And did you turn it into an aerosol for the fission product stuff?

MR. CAMPBELL: That I can't recall.

Casey, do you recall? I don't believe we looked at that.

MR. WAGNER: Yeah, Dave, at the time we didn't have the sodium fire models kind of connected to it. And that came up as there's quite a bit. And --

MEMBER PETTI: Oh yeah.

MR. WAGNER: And so that's something that now we would be able to do. And as a matter of fact, when we were kind of doing some vape ETR work, we kind of coupled in sodium fires in -- I think maybe Lucas might have slides on that.

I don't think we have anything right now for potassium burning, which is, you know, probably a hole that needs to be filled.

MEMBER PETTI: Thanks.

MR. WAGNER: Absolutely.

MEMBER ROBERTS: Comparing this to Steve Bajorek's presentation, he had two heat pipe designs he's evaluating, this one and the one he eventually like -- are you -- are you missing something? Or I guess the question for Steve, did you learn something from the second heat pipe design that would, you know, point to a gap here?

MR. CAMPBELL: On our end or on Steve's

end? I didn't -- if the question is directed towards us, we haven't looked at the eVinci design yet. That's something we still plan to do. The complication with that is having it in a horizontal geometry, right.

And so that's something that we're -- it's kind of the next phase. It's something that we want to be doing in the next year or two as to generalize this and allow for a horizontal heat pipe reactor.

But Steve, if you wanted to --

MR. BIELEN: This Andy Bielen, let me just say one thing real quick. One -- yeah, so I think, yeah, Volume 1 and Volume 3, our relationship and our collaboration has continued to kind of grow over the last five, six years, which has been really great.

I think one of the things that we learned from Volume 1, you know, they went and they were trying to build an eVinci-like model based on publicly available information. And frankly, as Steve alluded to, there's a reason that Westinghouse is planning on specific proprietary design features to make this thing work.

So you know, we're sitting here in Volume 3 saying okay, well, we have this gap we want to fill. You know, how are we going to do that. We looked over at the issues that, you know, Volume 1's having. You know, I was involved in that side as well.

And sort of like you know what, let's just put this on the back burner for now. Submittal schedule's a few years down the road, we want to be ready for it. But we also don't want to do a bunch of demo work that like we know isn't that applicable or there'd be big gaps that we would need to fill in anyway.

So I think that was -- that kind of helped us. That interaction and that collaboration helped us, you know, drive prioritization, I think. And Steve, you know.

MR. BAJOREK: Yeah, a couple, there's a few differences that you need to look at. When we did the special purpose reactor A, we didn't do it exactly the way they did it at the design in INL.

Because we wanted to change our set of oxide fuel, we wanted to go to a metallic fuel and a fast reactor. Because that was going to look much more like one of our -- one of potential applicants was going to be. So that exercised in a different way.

Now as you go to an eVinci-like, well, you have two things. You got a vertical orientation versus a horizontal orientation.

But also the way the fuel and the heat pipes interact in a -- in the metallic arrangement, the metallic fuel arrangement, the fuel could grow thermally away from the heat pipe. That creates another thermal resistance that you'd need to really account for and could be significant.

In the eVinci design, you're looking at rods and heat pipes in a graphite monolith. In that case, as that fuels heats up, expands into the model monolith, okay, actually improving some of your heat transfer. Course you're, you know, you have the horizontal behavior of the heat pipe, which orientation doesn't really -- orientation really doesn't matter a whole lot for the heat pipe.

Except one thing we did learn that in the vertical orientation, it's cooled off. You may put all of your sodium down below the evaporator. You're going to have a hard time -- you're going to have a hard time melting that when you want to heat up again. So there's -- each one has their own nuances to pay attention to.

MEMBER ROBERTS: Thank you, that makes sense.

MR. CAMPBELL: All right, if there's no

other questions, I'll move to the MSR. So for this one, we did the MSRE. It's a 10 megawatt thermal reactor, graphite moderated at near atmospheric temperature or pressures. Here the reactor is fueled with the dissolved fuel in the molten salt.

So 34-1/2 weight percent U-235. It has a really rapid transit time within the core. The 25 seconds roughly.

Some of the new modeling capabilities here. For SCALE, obviously modifications for handling liquid fuel. So for the nuclide inventory, we incorporated a time-dependent nuclide inventory to accommodate noble gas removal through the off-gas system, through the TRITON MSR addition, so it's a new module added. So we're able to model the time-dependent removal of nuclides from one mixture into another.

In MELCOR, we added thermal hydraulic equations of state for Flibe. We added a new model called the generalized radionuclide transport and retention model framework. And then molten salt chemistry and physics pertaining to radionuclide transport. And then we enhanced our fluid fuel point kinetics capabilities.

Accidents that we looked at for this one.

We looked at salt spills. We did it both in dry and wet conditions. In the wet case here, we assume a coincident water leak. So you get interaction of the molten salt with water on the floor.

However, in this design, there's a gas retention and then a condensing tank, which captured most of the radionuclides that are released from the spilled salt in those cases.

So some of our key insights here. You know, if you have your filter going, a filter fan going in the ops buildings, if it's operational, it's going to filter most of the airborne aerosols and you don't get a large release. But it has the other effect of also blowing xenon out into the environment.

And so you increase the release of the noble gasses, but you do decrease your aerosol release.

We had very few aerosol releases to the environment because of -- in all scenarios due to settling in the reactor cell, capturing the filter, or the salt spill case, capturing that condensing tank.

And then aerosol mass in the reactor building spanned many orders of magnitude depending upon your various scenario assumptions, so.

All right, and our final design. So we

have the sodium fast reactor. So for this one we did the ABTR. It's a 250 watt, megawatt thermal pool-type reactor using metallic uranium fuel with HT-9 cladding.

The reactor's fueled with those uranium, plutonium, and zirconium fuel slugs. Liquid sodium coolant, two pumps that circulate the sodium. And then it has four trains of DRACS.

New modeling capabilities for SCALE. Generating noble data for cartesian and hexagonal lattices and cells. New capabilities were added for that.

And then for MELCOR, we added material properties for sodium, metallic fuel, damage progression capabilities, and radionuclide release models. And as Casey mentioned just a minute ago, we've improved our sodium fire models.

The accidents that we looked at here were an unprotected transient overpower, an unprotected loss of flow, and then a single blocked assembly.

So for the UTOP, you have the highest worth rod withdrawals. Control rods fail to insert in the -- the -- and we did multiple sensitivities with varying reactivity insertions and saw -- looked at the fuel reactivity feedbacks.

Over here, this is the blocked fuel assembly scenario over here. And what we found that in a single blocked assembly, you got pretty rapid fuel melt, as you can see.

Here you can see the intact fuel. This is a single fuel rod going from intact fuel, heating up because of the drain of the sodium, going to solid debris, and then eventually molten in about 15 seconds.

So reality of this scenario is another topic. But in the case of a blocked assembly, we did see rapid fuel melt. And then here's the releases that we saw in that case.

So here I just wanted to talk a little bit about the V&V basis for MELCOR. Like I said before, we have a long history of code assessment dating back to the 80s and 90s. We're leveraging this assessment basis. We're moving forward from LWRs to non-LWRs.

So this figure on the right is trying to convey that a lot of the base physics that you can find in the modeling and simulation of LWRs is still present when we move into non-LWR modeling.

So for example, fission product aerosol release and transport is in most cases pretty equally applicable in both situations. And we have a really strong assessment base that we get to start from.

So from there, we have already conducted several code model assessments for a range of experiments that I have listed here at the bottom. But then also I show some of the assessments and benchmarks that we plan to do in the next year or two.

Also I show here some of the results of some of our assessments, including the IAEA CRP benchmark, HTTU and ATCOVE.

So for SCALE also, these are diagrams here for some of the assessments that are being done for SCALE at this time. SCALE's validation is broken up into four volumes.

So these are four volumes of validation documents that Oak Ridge is putting together for -- they have it broken up into four categories: spent nuclear fuel, reactor physics, shielding, and crit safety.

Here's three of the assessments that are currently being incorporated into the reactor physics validation case. I'll just point out that these are still being drafted. And this is kind of the next phase of our efforts in Volume 3, is performing these additional assessments.

But what I want to point is that we have done some and more are still coming.

MR. BIELEN: This is Andy Bielen. And just to further flush out, maybe get back to your point. Like, underneath the hood of these assessments is a database of data and models that are within a repo system.

We have access controls, quality control, all that sort of thing. So, like, we're making a big effort here to embrace modernity in our code development and make sure that we're able to really control both the things that we're doing and then the basis that kind of underpins that. And other thing I'll -- one other remark I'll make here is that we have some data that's available.

A lot of the assessment we've done thus far are taken from, like, the international reactor physics book and some other sources of data. Some of these concepts, either the data is legacy and it's not of great quality like MSRE. I don't think some of the data there was particularly -- given today's standards, things were a lot different back then.

Some of it was lost in the sands of time. So we are going to rely a lot on as reactors come online, as prototypes are built, we need -- I think one of the things that we pushed on, especially with our NRR colleagues and the vendors is we need support to help us, like, validate our codes with the same facilities that the vendors are using and building. So we haven't seen any big issues with that thus far.

MR. CAMPBELL: With that, I'll move into my summary slide. So what have we accomplished and where are we going? I hope you've seen that we've developed significant modeling capabilities for our SCALE MELCOR code over the last few years to address modeling gaps for the five primary advanced reactor types.

We've addressed modeling gaps through source code changes, model development, and even new work flows in our SCALE MELCOR codes. A great example of our code capabilities and readiness to support licensing was presented with the Hermes construction permit. In a very short time line, we were able to use our UCV Mark 1 model and apply it to the Hermes design to help focus NRR's review on safety significant aspects of the design.

Going forward, there's a lot of co-capabilities enhancements that we're still working on to improve our capabilities. Some of those are listed here. And then as has been mentioned many times as far as data needs, this is really the next phase of our efforts.

So we're always in need of more data, more assessment cases, more benchmarks that we can perform to make our codes more robust and ready. For scale, we could really use additional criticality and depletion benchmarks that are more representative of the fuel designs and conditions that we're going to see. And then for MELCOR, we need additional validation data on things like the diffusivity of fission products and varies fuels, heat and mass transfer characteristics in the diverse working fluids and so on.

But all in all, we do feel that SCALE MELCOR have been shown to be ready to support NRC's licensing reviews of non-light water reactors. So with that, that concludes my presentation. But I'm happy to take any further questions.

MEMBER MARTIN: I just one observation and of course, every slide is titled severe accident analysis. And you described your methodology through the referenced plans beginning with design basis events and then gingerly going into the domain of severe accidents. Is there a plan to kind of just dive in a little bit more and push these codes to truly the challenging what we consider severe accident limits, like I say, a next phase application of these models?

MR. CAMPBELL: We have done severe accidents in a lot of these cases, right? So if I show in all of these situations --

MEMBER MARTIN: Sure, like, the ABT -- like that one. That's where I said you start off in a DBA space and then you kind of do your sensitivities into it as opposed to designing events based on assessment of hazards.

MR. CAMPBELL: This is kind of hard because we don't want to get ahead of assuming what those cases are going to be, right? But we have explored a lot of these severe accident simulations. We have pushed the bounds in all of these workshops if you go and look.

We pushed the bounds into severe accidents in each one of these cases to, in many case, force a severe accident condition with fuel damage and release. And so in all of these cases, we haven't stayed just in DBA space. We have pushed the boundaries in all of these.

MEMBER MARTIN: Okay. But to the latter part of my question, is there any plan again to revisit more events? The models are there now. It should be easier. Is there any interest in the agency level to continue further in this area? Or are we considered done?

MR. CAMPBELL: I wouldn't say done. I'd say prioritization. We're really looking towards where's the priority of our efforts, right? Is it to go out and explore additional fuel melt accidents and break additional pipes.

Or is our focus instead to work on making what we have more robust and then seeing what industry is going to come in with? And they can best -- for example, let's say TerraPower comes in and has some novel accident. Then we can adjust accordingly versus being ready for every possible severe accident that could come about, if that makes sense.

MR. ESMAILI: Can I just jump in? I'm just going to make -- thanks, Shawn. So I think as Shawn said, at this point, we did the five workshops. And I just want to mention I think Dr. Petti said that the sources should be 50 times low.

So our emphasis is not looking at the numerical values. We were just trying to exercise the code because we have to break it to the point of getting something out. We have no -- so please do not look at those numerical values at all.

We just wanted to see what the sensitivity are. If I change this, how does this source term behave compared to this? So that's one point.

At this point, I think -- and this is my personal opinion is that to the extent possible, we have shown that what we have as we have done in the past in 12, 13 years ago when we were doing NGNP. We have the capabilities, right, to do model a lot of these accident sequences. And we do not have to do additional accident sequences with this model.

As Shawn said, we are convinced that we are ready to do this, these basic things. We need a little bit more validation on the modeling itself. And as we know a little bit more about the actual design, then we can go ahead and do this.

And again, as Kim said at the beginning, there's validation and verification. There is some we have a lot. Some places, we don't. So we just have to rely on a lot of uncertainty analysis, a lot of sensitivity analysis.

And if you look at the public workshops that we put in there, if you look at some of the cases, we looked at the sensitivity. Do I need to worry about the diffusivity of fission products? Or should I worry more about the particle failure?

So I need to worry about how -- what is the building? What's the issue with the building? Those are more -- has more to do with the source term than other things. So this is helping us to identify what is important, what parameters are important or not.

MR. BAJOREK: This is Steve. I just want to kind of add to that a little bit. I got to say that there are a few things that we should be looking at in terms of other accident scenarios.

And one I think we've talked about earlier, a steam generator tube rupture in a gas cooled reactor, one of the international benchmarks, they've identified that is the worst case. I forget exactly which one it is because of all the extra hydrogen you through into the system suddenly. It wasn't a scenario that we kind of considered early on because back in the NGNP days, I think the idea was not to have any water in the entire building.

Well, now we've got at least one applicant out there that's putting a Rankine cycle on there. So that's one that's new and different. And both Volumes 1 and Volume 3 need to start taking a look at that one.

If we're in a truly risk informed world, smaller leaks into the systems are something that we need to look at. We're used to looking at pipe breaks. And that's again a light water reactor legacy thought process.

But what about a small vessel breach in a molten salt reactor that is highly -- molten salt, it's highly corrosive, something like that? We should look at those now before we get the question in the middle of the review. So there are a number of things that we were planning on looking at in Volume 1.

CHAIR KIRCHNER: I think we need to stop here. We have more time scheduled for this afternoon. We've gone 20 minutes after the hour. How are we -- just calibrate, Kim. How are we in terms of your overall presentation plan? All we halfway, or --

MS. WEBBER: Yeah, yeah, we are. So in the schedule that we sent some time ago, in the afternoon, we have presentations on consequence analysis which is half and hour and then the licensing and siting dose assessment codes which is a little less than an hour. And then we also have a presentation on our fuel cycle analysis code. So I think we're about 15 minutes behind our initial schedule. So hopefully during the afternoon, we can make up a little bit of time.

CHAIR KIRCHNER: Okay. Well, we'll have a hard stop later this afternoon at approximately 3:15.

MS. WEBBER: 3:15? Okay.

CHAIR KIRCHNER: Let's reconvene at -- let's see. Can we take a whole hour here? Yeah, let's reconvene at ten minutes after 1:00.

(Whereupon, the above-entitled matter went off the record at 12:24 p.m. and resumed at 1:10 p.m.)

CHAIR KIRCHNER: Okay. We're back in session. I'll turn back to Bob Martin.

MEMBER MARTIN: Okay. And I'll probably just turn it back to Kim to introduce the second part --

MS. WEBBER: Yeah.

MEMBER MARTIN: -- of Volume 3 --

MS. WEBBER: Yeah.

MEMBER MARTIN: -- of consequence analysis.

MS. WEBBER: Okay. Today Luis Betancourt is here to represent our successes in the MACCS and consequence analysis area with one of his senior staff members, Keith Compton. They're both in the accident analysis branch. And so let me turn it over to Luis and then Keith.

MR. BETANCOURT: Yeah, okay. Well, good afternoon. Thank you for allowing us to speak today after lunch. So I hope you guys are happy. So feel free to ask us any questions along the way.

So as Kim mentioned, my name is Luis Betancourt. I'm the branch chief of the accident analysis branch with Pyrra M. Tudesky come to now I'm a senior reactor scientist. And we wanted to discuss today is regarding what is a success story that we have been at this time, readiness activities for the MACCS consequence analysis computer code.

You're going to be hearing the presentation today is basically kind of the key answers that we are going to be as ready as we can be at this time. And one of the things that we're going to be focusing more in the next couple of years is to work the SSIs so they can code. You guys heard a lot of the messaging today about readiness, that we need to be able to build expertise in house.

So you're going to be hearing some of that in the presentation. So I'll turn it over to Kim to discuss the slides. I'll turn it to Slide No. 2.

MR. COMPTON: Can you hear me? And everyone -- okay, it sounds like I'm coming in the microphone. So good afternoon. I'm Keith Compton as we said.

So the first thing that I want to start off, I'm going to start off with what I think of is as our key messages. And then the first key message is that we basically expected to wrap up most and possibly all of the tasks that we identified in the code development plan by the end of this fiscal year. And I'll get into what that means.

It doesn't mean that we're going to stop working at the end of this year. I'll talk a little bit about that. The approach that we've been taking throughout this process is that we were looking to see whether there was an identifiable code improvement, a MACCS code improvement that could address the topic that we were looking at that was consistent with state of practice.

And the concept of state of state of practice is something that I'll kind of go through a lot. We're not trying to go beyond state of practice. And in some cases, we adopted algorithms from state other state of practice codes. And other cases, we recognize that there was nothing that would represent a substantial improvement over what MACCS already does.

So that's kind of a philosophical approach to how we address our code development. We also concluded that the motivation for some of the tasks that are in the code development plan were predicated on a hypothetical but an unspecified difference in the physical and chemical forms of release radioactivity relative to -- for advanced reactors relative to existing light water reactors. And that's something that I'll pick up again later that's important because we are finding that there are a number of codes that address unique physical and chemical forms.

They're typically highly specific to specific physical and chemical forms. Some examples would be tritium is unique. They are dedicated for the codes.

Another example that you're probably all familiar with is UF6. There are dedicated codes that handle UF6. What we're not finding is that there's kind of general purpose codes that handle anything that you might through at it.

And that affects part of our planning. So basically, yeah, we can't -- we're not going to try to keep the plant open to handle every possible form that may be encountered. Next slide, please. So on this slide, I'm going to -- I'm just going to briefly talk about the status of individual tasks.

And I would note that we have supplemental slides that give more details for each of these topical areas. So for near fuel modeling, we benchmark the MACCS against several state of practice dispersion codes such as AERMOD and QUICK and ARCON96 to look at the performance in the near field. And the bottom line is that we identified some algorithms that we could incorporate into MACCS and we incorporated them in MACCS. I'm not going to talk about any of these details unless someone wants to pull off -- wants to go into any one of these areas. So for the next task --

CHAIR KIRCHNER: Keith, not to slow you down. You're on a roll.

MS. WEBBER: That's because I told him to go fast --

(Simultaneous speaking.)

MS. WEBBER: -- to meet your schedule.

CHAIR KIRCHNER: -- all of a sudden becomes a lot more important as we see applicants trying to bring in their exclusionary boundaries, bring in their LPZ, et cetera, et cetera, or bring in the EPZ planning zone and so on. So could you just spend a little more time? So is it ARCON92 or whatever -- 96, is that your work horse for adjusting the near field?

MR. COMPTON: Sure. Let me get into it. So the issue with near field is that the MACCS as traditionally used was kind of configured and typically parameterized to handle offsite distances of more than about 500 meters. The significance of that is typically by and large beyond the wake effect of buildings.

And there was actually just a typical parameter that was used or it was an approximation that was only valid at out to the 500 meters. So what it is we looked into how could we have MACCS -- what are ways to use MACCS closer in? And we identified. You can actually -- you could've used the existing MACCS in a very conservative way just by assuming your source is a point source.

(Simultaneous speaking.)

CHAIR KIRCHNER: Excuse me. But do we have people with open mics out there on Teams? Please mute yourself.

MR. COMPTON: All right. I was going to see whether I can keep the pace going. So right. So we looked at the algorithms for have a typical state of practice codes to do the near field dispersion, the dispersion kind of in the hundreds of meters range.

The short answer is that we found an algorithm that was used to develop ARCON96, the Ramsdell-Fosmire model.

It was based on fuel students that were done in the vicinity of nuclear power plants. I think the fuel students were in the '70s and early '80s. And that algorithm for accounting for the enhances dispersion from wakes was actually more important.

The enhanced dispersion due to low wind speed through the air and -- that's a general applicability. So we were able to look at the technical basis of the equations, incorporate those into MACCS. We then compare that to see, does it give comparable answers to ARCON96?

And then we also compared to AERMOD, EPA Workhorse code. Then we determine that we believe that it's a suitable way to have dispersion estimates at close ranges that are appropriate but not overly conservative in a the way that's simply using a point source model with no enhancement for dispersion, no enhancement for meander. That would be conservative and that could work, but it could be overly conservative. Does that help? Okay.

So for the next task, the radionuclide screening, we reexamined the technical basis for the original list of 60 radionuclides that are generally considered for light water reactors. And this is one of the first examples of where one of the real benefits of this work is that it forced us to go back and make sure that we understood not just what we could do for non-light water reactors but why are we doing what we currently do for light water reactors. So we did that.

We reexamined it and we essentially came up with a quantitative methodology for selecting radionuclides, screening radionuclides that use the same considerations that we use back in -- well, since WASH-1400 actually. So the half life, the radiological hazard, the abundance in the core. So that task, we kind of considered we're done in the sense that we've identified how to do.

Of course, you're never done until you know the inventory. And that could always change. So this is an example of where we're done but there will always be more work to do.

We figure out how one could do it then. So the subsequent task, we examine whether there were state of practice methods to address the effect of variability and physical and chemical forms on dosimetry and atmospheric dispersion -- and I'm sorry, atmospheric dispersion and deposition. And we concluded that MACCS capabilities were broadly consistent with state of practice.

Again, this involved going back to understanding why did we pick the chemical forms, for example, for dosimetry, federal guidance report 13 which is one of the standard references for dosimetry for radionuclides -- for environmental exposure to radionuclides. There are multiple chemical forms that you can assume for radionuclides. But it's somewhat constrained.

You're limited to what the dosimetrist have assessed. So we went back and we looked at how did we pick what we originally picked. So we realized that MACCS is basically a state of practice.

It has the ability to -- you can change the dose coefficient file, for example. You don't need to do a code change. But you need some more -- you need to be more conscious about not just using defaults without thinking about whether it's appropriate for your application.

But again, we don't think that's a code development issue. We think it's an understanding your own source term issue. So for examining the consequences of tritium releases, we benchmarked MACCS against two state of practice codes.

And we determined -- and the state of practice codes were UFOTRI and ETMOD which is a Canadian code for tritium releases. And this is consistent with observations that the DOE has made. Tritium is an issue for DOE facilities that MACCS can be used for evaluating inhalation doses from airborne tritium.

Basically, it can be somewhat conservative or it can be fairly accurate. We did conclude that MACCS is not suitable for estimating ingestion doses from Tritium. That's a different pathway.

But a solution to that, the question -- and this is the debate or the discussion we're having. Do you then upgrade the MACCS code to put in that special purpose capability? Or do you simply -- and this is the approach that I believe has been taken in other applications. Do you simply use a special purpose code if you have to do that specific task?

So I'm kind of leaning towards the direction that -- and the other is understanding the risk significance, the dose significance tritium can -- you have to release a large amount of it to get significant doses. It's possible, but you should be thinking about whether you're putting in a lot of capabilities that is going to end up not being of great significance. So we got into the literature and got a little bit more informed about how to help guide that decision.

So the remaining tasks, I'm going to leave with the remaining tasks are not complete. But we're considering closing out the final -- two of the final tasks in the co-development plan without extensive work on it. And I'll talk about why on the next slide. So next slide, please.

MEMBER MARTIN: Going back to your second bullet, is there some sort of guidance that you put out to either applicants or at NRR as to how to do the screening study or what the expectations are to screen out the radionuclides that are important to dose?

MR. COMPTON: So the second -- it's just the bottom report gives -- explains the methodology and talks about how to do it. I want to be careful about saying if we've given out guidance, I would say that's something for the program offices to figure out what they want to say about guidance. But we did put out a report that we believe explains it in sufficient detail.

Someone could pick it up and understand how one would go about doing that and how it applied to. And they can take that approach and apply it to different inventory. So we tested it on an example.

MR. BETANCOURT: I was going to say that. So basically like a -- you will see that in the next couple of slides. So we have a demonstration project that we basically use and a sample source to demonstrate how comparability could be used. So just give us a couple of minutes and then we can talk about that.

MEMBER PETTI: Can I go back to the tritium issue? This is really for accident.

MR. COMPTON: Right.

MEMBER PETTI: For Part 20 evaluations, are the codes good enough? Do they have the skin absorption do you know for the chronic release stuff?

MR. COMPTON: I won't talk -- I'll answer that of the way. I won't talk about Part 20. What I will talk about is MACCS. So one of the things that I think we are going to do, I can't remember if we've done it yet or not, the typical approach for skin absorption is to increase the dose coefficient by a factor of 50 percent, the inhalation to this coefficient to account for the enhanced dose and skin absorption. And I think we're going to make that change so that MACCs will -- dose coefficient file will ship with that increase.

MEMBER PETTI: Because at least data that I've looked at, I mean, HTO versus HT, huge difference.

MR. COMPTON: It's huge. And the dose coefficient -- again, the -- I hate to say default because you should always know why you're using the code. But what I would recommend would be using the dose coefficient commensurate with HTO. And then --

MEMBER PETTI: That's not what's happening. That's what I'm worried about.

MR. COMPTON: Well, and --

MEMBER PETTI: In some applications, they're coming out at HT. And I can justify that.

MR. COMPTON: And in MACCS, we do have that. And that's why I won't speak to what others do. And that is important. And that is part of what we found in the benchmarking studies is conversion of HT comes out as HT and everything is fine until it converts, until the hydrogen -- or the tritium gas, HT, coverts into tritiated vapor. And then suddenly it's much more bioavailable and you see that when you run something like this track. So our answer is just we'll just assume that it's HTO to begin with.

MEMBER PETTI: UFOTRI I always considered to be a benchmark.

MR. COMPTON: Right, right. And I agree with that. Again, to go to Luis' point, one of the things that has been beneficial is that we're getting staff having the experience of not just running MACCS but running UFOTRI and talking to Dr. Raskall.

(Simultaneous speaking.)

MR. COMPTON: So yes, so this -- getting to know the literature. And again, it's interesting. There's not a lot of people that run UFOTRI. There's not a lot of people that run that. So it's important to --

MEMBER PETTI: Right.

MR. COMPTON: This exercise was important. Even if we don't change the code, we understand how to use the code.

MR. BETANCOURT: And one thing before we're moving on, I think this is what the international collaboration through certain organizations has been very beneficial. And we've been having those exchanges. So that's something that we need to continue doing.

MR. COMPTON: I will say that I was happy when Dr. Raskall came to the presentation that we gave. Our contractor gave the presentation and we did not get jeers or whistles from Dr. Raskall. But I felt very happy about that.

So let's see. Okay. So that was a brief status of where we are and where we expect to be by the end of the year. I'd emphasize the fact that we plan to wrap up the work identified as part of the co-development plan does not mean we're going to stop working on MACCS and stop looking at things. It's simply while recognizing that we may be doing focus work in the future.

We think that MACCS is ready for use in assessing the offsite consequences for a wide variety of non-light water reactor technologies. So we're reasonably ready is what I would say. And I think we can deal with when it comes up on a more case by case basis.

And one thing that is worth noting is that MACCS was already upgraded in the 1990s. One of the motivations for the development of MACCS too was to support application to DOE non-reactor facilities. So it already had a large amount of technology neutrality kind of baked into it.

So we kind of -- we're already starting from a good spot. And one of the things as we've always tried to do, we plan to stay abreast of source term development work to see whether there are specific MACCS enhancements that are needed. So I think that we're pivoting now more towards kind of pay attention to what may be coming to be assessed with MACCS and looking at the specifics instead of trying to solve problems on a generic basis.

And one other thing that I would mention is that we need to make sure that any work we do has a clear nexus to a regulatory application for which MACCS is suitable and for which we have the requisite technical expertise. And the reason that I mention that is that we have a task identified on chemical hazards. But before we charge off -- and that's something MACCS was already -- in the 1990s, there was a code called KIMACCS, an adaptation that was done that used MACCS for assessing chemical hazards. Before we charge off and start -- bring that up, resurrect it, build in all these capabilities, we need to make sure that we're the right people to be doing that and that we're solving the problems that are -- so that's, I think, a discussion that we're going to be having with the program offices.

MEMBER PETTI: So one other question is here. I understand it's not this one, right. But from a health physics perspective, I don't know the answer. Is there some sort of synergistic enhancement when you've got chemical release and radioactive release at the same time?

Could be in the same particle like the public would inhale. I think of plutonium as a classic, right? It's got its radioactive stuff. But it's also toxic. There are materials that could be co-released, and I don't know if that's captured at all.

MR. TOMON: So in the code like the RASCAL code, that is captured for, like, HF. It is captured in there. It does both, the chemical and -- it's a very infrequently exercised portion of the RASCAL code.

And we're actually looking in modernization to keep it in RASCAL or move those DLL files into some of our other codes. But you're right. But we have some codes that do look at it that way.

And then it used it be that we used to have a few separate codes with RADTRAD and HABIT. And we still have the two separate codes. But with HABIT, we've completely taken out the radiological aspect because it's for design basis accidents in accident space. And that's what RADTRAD has bee upgraded through the years to do in SNAP/RADTRAD. And so HABIT code is strictly for chemical and eventually when we get to my presentation, that's one of the steps that we're -- because we have so many codes is to combine HABIT and RADTRAD into the same user interface and then bring in some of those additional code interactions between both the chemical and radiological.

MR. COMPTON: And one thing, though, that's an example of a specific question that is actionable for research. That question of -- because in general, I mean, that's something that I think the EPA deals with, with chemical hazards is things can be -- hazards can be additives. They can be synergistic or they can be antagonistic.

Sometimes one can kind of cancel out the other. So it's a -- generically, it's an issue that has been identified and addressed. And I personally would approach because I have a strong bias towards staying within the state of practices, understanding what is a consensus state of practice method for dealing with that kind of issue. But that's a very specific -- I'm glad you posed it that way because that's something that we could look at and figure out how to address it.

MEMBER PETTI: And the other thing is in

terms of the isotopes that you're considering, I'm assuming you've also looked at activation of all the coolants and the moderators that are out there. It's not just fission products. There's some sodium that has isotopes that activate that can give dose potassium. Some of the solid moderators that move beyond graphite and some of these micro-reactors will produce tritium and other things that just make sure it's in the --

(Simultaneous speaking.)

MR. COMPTON: And that's an example of why -- yes, and we've always, in theory, looked at activation products. But we would rely -- I would say we would rely on the output of the scale and origin calculations which include not just the fission products and the transuranics. But it has -- but we had -- right.

MEMBER PETTI: Yeah, because sometimes, like, the moderator is separate from the fuel in some cases. So the scale calculation might only be on the fuel and not look at activation in the moderator. They'd have to know.

MR. COMPTON: They would have to know. And I think that's the -- and that, again, gets to the point of we need to talk to each other because something which might not be terribly important from a neutronics point of view, for example, might be significant from a consequence point of view. But in general, again, we have a generic methodology that if you know the activation product inventory, you can figure out whether it ranks -- it should be included in your -- and MACCS can handle any isotope. Well, it can handle pretty much any isotope that is for which you can have a dose coefficient, FTR-13 which is 825.

MS. WEBBER: Hey, Keith. If it's okay, I'm going to speed you up a little bit.

MR. COMPTON: Sure. Well, okay. So the next slides I think will be quicker. So next slide. So like I said, the conclusion that we're ready for use in assessing offsite consequences doesn't mean that we're done.

There are several candidates for future work that identify. But they're applicable to both light water reactors and non-light water reactors. So I think those kinds of things, we move them into just kind of our normal development efforts.

Keep them siloed in the non-light water reactor issue. Just improvements that we can make. I'm not going to go into these listed in detail. But these are just some specific examples of things that we identified over the course of these exercises.

Again, I make the point that our co-development plans have been -- our development work has been very useful, not just in looking at our code but our knowledge management. It's given younger staff the chance to dig into things that they otherwise wouldn't have an opportunity to dig into like the tritium work. Like, the work on understanding why we use the chemical forms that we do currently.

And that's been very helpful. You don't usually get an opportunity to go back and do that kind of work. So let me see. Where -- okay, thank you. Losing track of where I am.

One of the things I won't go into detail but an activity that we've been working on for advanced reactor readiness, it's not explicitly listed in the co-development plan. But it's something that we identify that we needed to do. And we just basically realize you learn by doing.

You learn by actually trying to do the assessment. So as an example, we took a MELCOR source term that was generated for the source term demonstration project. And we just said, let's run it through MACCS and see what it takes to get that done.

I won't go into the details. But I will say, yeah, we learned some things. We learned some things that we probably wouldn't have identified if we had just tried to sit back and guess what it went into.

The mechanics of developing inventory, the mechanics of coupling the source terms, the issues of, oh, what happens if you have a transient overpower where your reactor is not scrammed and you're having releases. MACCS kind of assumes that you have a scram and then you have your release. Well, what do we do?

So that's useful. It's useful to actually do things and not just kind of speculate about what you might need to do. So I think that's the kind of the thing that we'd like to keep doing going forward is to keep practicing as it were, practice on different kinds of source terms and see what we learn. And then also again, that leads so -- that's a chance for staff, both at the NRC and contractor staff, to learn how to use the code. And I think --

CHAIR KIRCHNER: Just to kind of see if I can make this a general question but specific enough. You've got the potential for energetics. Is that a MELCOR responsibility to give you an energetic propelled release? Or is that something you take up in MACCS and try to account for it?

I'm just thinking of the fact that you could have either forged materials. You can have extra energy sources. How does that impact the dispersion --

MR. COMPTON: Right, okay.

CHAIR KIRCHNER: -- aspects?

MR. COMPTON: That's an interesting question. So I'll give some thoughts on it. So there's a couple of ways that energetics could come in. My reaction, the first thing is whether you're modeling a release as a buoyant plume or as a jet.

So we're aware of the fact that we tend to model things with buoyancy and the momentum dominated effect from a jet. I think we believe that we typically assume is dissipate fairly quickly. Again, that may be kind of an airfield long-term issue.

In that case, I'll also say the orientation of the jet matters. Is it going up? Is it going down? So my answer to that is probably unsatisfactory.

I think again it goes to the importance of understanding and again making sure the staff continues to understand -- new staff coming up, understands what the assumptions are that MACCS has in it and then figure out whether it's applicable. So I will say that the question of -- well, I would say that we could not -- that would not be in our swim lane to figure out whether it's -- what the energetics are, whether something more like an exposure or more like a low velocity release. We'd have to understand that from I think the source term development.

I would say again -- and this is the importance of staying on top of what is going on in other areas. There are people who work in dispersion codes where that is an issue. And I think having our staff stay on top of what's going on in other areas and what is state of practice for modeling burst releases or vent releases or jet releases, to figure out whether do we need to make a change or not.

CHAIR KIRCHNER: It would likely be very application specific actually, depending on the chemicals and/or pressures, other factors in the accidents.

MR. COMPTON: Right. And Luis will probably pick up on this. But again, the feeling that I'm taking up is that we -- it's reiterating we need to get out of our -- we need to make sure we don't stay in a silo. We need to talk to the source term people.

We need to talk to other communities of practice that learn from those so we're not reinventing the wheel so that we're conveying you need to give us this information so that we know what we need to do with it. So kind of a meta level, that one nation gets highlighted when you start doing something a little different than what you've been doing for the last 20 years. You got to talk to each other.

MR. BETANCOURT: Let me wrap this up so we can actually go to the next presentation. So I think what you guys heard today that at least for MACCS it tends to be more like a technology agnostic code. And I think we are as ready as we can be from a co-development practice of Volume 3.

And our plans as Steve mentioned is just to continue some of the case-by-case basis that's out of our standard co-development activities. And one of the focus areas that I would like to do, at least in fiscal year 24 and beyond is to do more of the exercising of the code, use some of the source term calculations just to get some insight to basically help and build that expertise in house. And that's all that I have, just to keep it quick and simple. So any questions or comments before I turn it over to the other gentlemen? Go ahead, Vicki.

MEMBER BIER: Yeah, I have a more general question that might be fore Kim or Shawn or somebody who presented earlier. I mean, you said, like, hey, we're as ready as we can be, right, which sounds pretty good. And Shawn's presentation, he said, well, there's some areas where we really can't do much yet because we don't have the suitable data or whatever.

I guess the question that I have, like I said, it's probably better for Kim who's sitting behind me. But are there areas where you guys collectively think there are critical development tasks that you have the data and analysis to be able to do NR cost constrain? Or do you have the -- is the --

(Simultaneous speaking.)

MS. WEBBER: I'm going to address that in my conclusion. I'm going to address in my conclusion. Okay. But you're leading me down the right path.

MEMBER BIER: Thank you.

MS. WEBBER: Would you like me to introduce the next speaker or do you have anymore questions for Luis and Keith?

MEMBER MARTIN: Just real quick. Obviously, with the risk informed licensing coming along, frequency consequence is a big focus of that.

And I think that's going to mean more attention on what you do. And it seems that your conclusion is, well, we're ready. Maybe there's still question marks on how people will use your code in that framework that you're seeking answers to.

I'm curious if you look at something like near field, dispersion as a physics model that maybe you could improve on, what have you. But you have questions about how people are going to use your code down the road because they are that you can maybe prognosticate on now and add to this -- future is kind of nebulous because obviously there's no commitment to it. But I'm just a little concerned that we're not going to be ready when people are starting to be creative with the presentation of frequency consequence at the level that I think it's going to get to in maybe the next five years.

I just seems like there's a lot hanging out there, questions. Again, maybe you don't know until people try those out. But I do think you're going to get more users, and they're going to push it in ways that you maybe haven't thought of or want more modeling capability.

MS. WEBBER: Can I jump in on that one? Generically, we have the capability to look at a lot of things. But when it comes to very specific detailed things as you're seeing, we just haven't got the budgeted resources. We don't have the time yet to do a lot of these very specific things.

But doing the code analysis is not the only thing that the NRC has to really assess safety and protection of the environment and all that. I mean, we have regulatory tools to address key areas of uncertainty beyond doing confirmatory analysis. So you can put limiting conditions on operations.

And so collectively, we think that with what we have today plus the regulatory tools that we have that we'll be able to do some of -- do this as safely as we possibly can. But we're you're seeing is we've been working on this for a number of years. It kind of goes to Vicki's question too.

We've been working on this for a number of years. And with more budget and more time, we'll advance our capabilities even further to address some of the really important questions that you've asked about today. So yeah.

MEMBER MARTIN: And just the intention is obviously going to not be on fuel temperatures as much as it's going to be on the dose. And people are going to start to want margins in that area where maybe in the past we're not mechanistic or what have you. And maybe we're looking five to ten years out and not near and that's the next vision statement.

But I do think that's the trend. I think more and more people are going to be happy with the risk informed approach that focuses on radiological safety issues. And you may find more demands coming because of the changes. I think the landscape is changing and it's going to be focusing more in this direction.

MR. BETANCOURT: And what I will say about that, I've been in this position for almost three to four years already. And I've been seeing an increase on MACCS users throughout the years. So we have been seeing, I will say our numbers have increased 100 or more, give or take.

And we're keeping tabs on who's using it for what reason. We're trying to keep tabs also on what the MELCOR users are also doing to the point of watching them. So it is that we want to figure out how people are doing it.

Do we have a cap? I think that's one of the ideas that we're not learn by doing. But the point that I was trying to make is that we are as ready as we can be from a generic sense. Now when you're talking about case-by-case basis, that's when we need to be able to following the standard process of co-development activities.

And at that time, hopefully we'll be more proactive. We identify decisions before they come along. But there has been an increase of MACCS users throughout the years since I have been on this job for three or four years.

MEMBER MARTIN: Is there a formal users group?

MR. BETANCOURT: Internal. Yes.

MEMBER MARTIN: Okay. But not outside of the broader --

MR. BETANCOURT: Correct.

MEMBER MARTIN: Okay, yeah. All right.

MEMBER ROBERTS: And I can't resist the opportunity to put in this plug again. I think that what you just mentioned is an example. That's going to put -- I think it's going to put more onus on not the code but the code user, the analyst.

I started off my career in performance assessments, waste management where the first thing you do is to figure out what is the right tool to use. You don't just kind of go to your standard tool. Am I doing a granular problem? Am I doing a surface water problem? What kind of problem is it?

And so I think that puts more new technologies, new scenarios means that you need to be a little bit more conscious of am I using the right tool? Do I have all the -- am I looking at all the phenomena that I need to look at? What's useful to us to understand things like, well, is this applicable? I would say MACCS model models neutral density aerosols.

It's not a heavy gas code. It's not a lighter than air gas code. It models generally non-reactive, so things that don't undergo complex transformations.

So having the experience to know what is my code designed for, how well does it suit the problem that I'm trying to solve is important. You can't just kind of go, oh, I'm going to use what I've always used and how that it actually applies. So I point out this is the reason that I think we need to emphasize not just the codes but also the knowledge management and the skill development that you can -- you can't put everything into the code. So that's why I think the code development, trying to learn by doing is important.

MR. COMPTON: Move on.

MS. WEBBER: Yes. So let me introduce John Tomon. He's the chief of the radiation protection branch. And I'm going to ask John to try to go for 30 minutes --

MR. TOMON: I will try.

MS. WEBBER: -- acknowledging that there'll be questions along the way.

MR. TOMON: I'll try.

MS. WEBBER: Thank you.

MR. TOMON: I have a script, so I don't know how long it is. I haven't timed myself. So I'll try to keep to the high points and go from there.

As Kim said, my name is John Tomon. I'm chief of the radiation protection branch in the Division of System Analysis. This afternoon, I'm going to provide an overview and update on the Volume 4 license and siting dose assessment code to the activities we've done.

In the following presentation, I'll discuss the work my staff in collaboration with our contractor Pacific Northwest National Laboratory have undertaken and completed with respect to the task developed in Volume 4. Next slide, please. This next slide -- I'm trying to go fast. So Volume 4 describes a vision strategy to achieve readiness with a non-light water reactor designs for the licensing and siting dose assessment codes.

The staff and the code contractors identified several issues within the current suite of licensing and siting dose assessment codes which should be addressed in preparation for all non-light water reactor technologies as well as maintaining their applicability to the current light water reactor fleet. Working with our individual dose assessment code developers and the radiation protection computer code analysis and maintenance program contractor, Pacific Northwest National Lab. And for those of you that don't know, the RAMP program or radiation protection code analysis maintenance program is our cooperative code sharing program, both internationally and domestically with licensees users.

We have over 2,800 users in RAMP because of the amount of code that's in RAMP. The staff developed a five test listed on this slide to prepare the licensee and siting dose assessment codes for non-light water reactor readiness. These include looking at code consolidation and modernization, improved characterization of source terms, improved atmospheric transport and dispersion modeling, update dose coefficient values, and update to the environmental pathway modeling used in some of the codes and where necessary, include pathways which may be important for non-light water reactor technologies. Next slide, please.

As shown in this image from Volume 4, we're looking towards the possibility of having to make approximately 10 licensing and siting dose assessment codes for the various non-light -- ready for the various non-light reactor applications. These include codes like the atmospheric relative concentrations code in support of control room habitability, ARCON, you've already heard about it, the ground level relative air concentration code for accidental releases, PAVAN, the gaseous and liquid effluent release code, the normal affluent dose assessment and siting code, NRC Dose 3, which includes liquid pathway modeling dose assessment code, LADTP, and the gaseous and atmospheric pathway modeling dose assessment code, GASPAR.

NRC Dose 3 also includes the normal relative air concentrations and relative disposition factors code, XOQDOQ. These also include the radioactive transport removal and estimation code which has access via the symbolic nuclear analysis package model letter/number. We refer to that code as

SNAP/RADTRAD.

And then finally, the control room habitability code which we've already briefly mentioned. The code graphic on the slide shows the various does assessment computer codes in RAMP. Currently there are 20. In a few slides, I will show you a revised graphic of the consolidation of the licensing and siting codes.

Also in Volume 4, we also included discussions on other RAMI computer codes that either non-light water reactor designers are considering using in their applications. And we know this through the interactions of our RAMP user group and our RAMP user meetings. These are codes such as the radioactive material transport dose assessment code, NRC RADTRAD, the Generation 2 code called GENI, and the decommissioning codes which we're not really working on right now because we feel that will be later on, further on. So we pushed that out to a later phase in development and as we work on the licensing and siting codes. Next slide, please.

This slide depicts the current licensing and siting dose assessment codes that the NRC staff uses to perform independent assessments and confirmatory calculations with respect to the regulations in various parts of the code of federal regulations and the NRC regulatory guides for light water reactors. In Volume 4, we group these dose assessment codes in areas of licensing reviews based on the source terms and the type of reviews the codes are used for. As shown in the graphic, the GALE code with its four subroutines along with the NRC Dose 3 computer code with its three subroutines are used together to calculate the dose from the normal effluent releases from light water reactors.

Additionally, the relative air concentration outputs from the ARCON and PAVAN computer codes are used as inputs to the SNAP/RAD computer code to calculate the dose to the control room, low population zone, and exclusionary boundary for design basis accidents for light water reactors. And then finally, HABIT code is a suite of codes to assist evaluating light water reactor control room habitability in the event of accidental spills. Next slide, please. This slide depicts the future of the licensing and siting dose assessment computer codes after the code consolidation and modernization.

As shown on this slide, the consolidated licensing and siting dose assessment code called the software integration or environmental radiological release assessments -- from now on, I'll just refer to that as SIERRA because it's a mouthful. And I almost referred to it to begin with because it is a mouthful. We'll replace the computer codes used to calculate the doses for normal effluent releases from existing light water reactors and future non-light water reactor designs.

Likewise, the RADTRAD computer -- SNAP/RADTRAD computer code will be combined with the HABIT code to assess control room habitability in the event of accidental spills of toxic chemicals and accidental releases of radionuclides. Finally, the atmospheric transport and dispersion computer codes, ARCON, PAVAN, and XOQ over DOQ uses similar calcium plume model. The decision was made by the staff and our contractor to consolidate all three into the SIERRA code atmospheric transport and dispersion module with the output, the relative air concentrations for the near field which would be the ARCON type calculations and the midfield, the PAVAN type calculations, to be readily imported into SNAP/RADTRAD for design basis calculations at the control room and the low population zone and exclusionary boundary. Next slide, please.

This slide shows the accomplishments we've done to date based on those tasks. The staff with our contractor, PNNL, have completed the task listed on the slide from Volume 4. The code consolidation and modernization work began with the development of the consolidated -- consolidation -- the code consolidation framework in late 2021.

As mentioned earlier in the presentation, this consolidated licensing and siting code is referred to as the SIERRA code. We'll combine individual FORTRAN codes into the one that will pass data quickly and efficiently to the various modules in SIERRA. The figure on the right shows all the codes under the RAMP program with the SIERRA code consolidation entered at the top of the code wheel and the HABIT code included under the SNAP/RADTRAD computer code.

The second completed task is a Phase 1 work to improve characterization of the source term and the SIERRA computer code. Specifically, Phase 1 for the source code -- for the source term involves the incorporation of the existing light water reactor, normal reactor cooling source term computer codes, the GALE -- the four GALE subroutines into the SIERRA source code module. Lastly, the third completed task is improving the atmospheric transport and dispersion models which includes consolidating ARCON, PAVAN, and XOQ computer codes into the atmospheric transport and dispersion model.

And then the next few slides, I'll go over those completed tasks in a little bit more detail. Next slide, please. Code consolidation approach, one of the first identified priorities of code consolidation was increasing the efficiency and maintaining the large numbers of licensing and siting dose assessment codes and preparing for the different types of non-light water reactor designs and fuel types being considered with the resources available. Code consolidation and modernization was viewed as a means to help remove functional redundancy between codes, improved outdated science and technology associated with design and development of the original codes, the legacy codes, address limited ability of the current codes to assess advanced reactor designs, apply a standard software quality assurance to address a history of changing ownership and associated lost of code development and knowledge, and finally, reduce the inefficiency of having to maintain multiple codes.

The figure on this slide is a diagram of the consolidated code paradigm showing how the models from the existing or legacy licensing and siting codes are going to be integrated into the new SIERRA code. The modules within the SIERRA code are grouped or characterized within the general dose assessment approach. The SIERRA code has eight modules as shown on this slide which will contain similar phenomenological models with the current licensing and siting dose assessment codes. Next slide, please.

In order to address the challenges identified for the current suit of legacy codes, a three pillar approach was adopted for the SIERRA code which includes the following steps, first creating consolidated engines. This is a set of functional models or engines that are being developed to perform regulatory calculations as those performed by the current suite of licensing and siting codes. In most cases, we're going to bring in what we already have and then build upon that is what the plan is as we work through this.

The functional engine approach improves development flexibility by allowing for future modifications and efficient data transfer. Furthermore, separating these capabilities as standalone modules eliminates some of the current code redundancy and inefficiencies. The second was to develop -- the second pillar was to develop a standardized data transfer schema using a standardized data transfer, JavaScript object notation or JSON for encoding data for each engine makes the data input universal and adaptable while making it easy to pass the output data between the different functional engines.

By using JSON as the data transfer file format within the SIERRA code framework, the entire system is more robust relative to the advancements in the nuclear industry and any associated improvements in data entry such as downloading logical input data. Finally, the last pillar was to built a single user interface. The single user interface has already been developed, separate from the functional engines which acts with the users and communicates with the functional engines to execute user defined commands. The user interface is designed to effortlessly guide users through the relevant code engines input screens.

This approach allows for an adaptable code that can consolidate functions of the existing codes which were bringing in a lot because many of these codes have been updated recently for growth and expansion for new challenges as they arise. Next slide, please. The figure on this slide gives you an overview of that graphical user interface showing how the slide -- how the user can access any of the functional engines in the SIERRA computer code. The SIERRA code atmospheric transport and dispersion module has really just been completed -- has just completed beta testing with the meteorologist from the Office of Nuclear Reactor Regulation.

And then the contractor is taking their feedback, suggested edits and bug fixes which were detected during this review. And the staff anticipates releasing the SIERRA code completely to the user community with the combined atmospheric transport and dispersion module at the end of September of this year. I have a few more slides that show a little bit more about that combined module for Task 3 and subsequent. So if you have any questions about the ATD module, you might want to wait until we get to those slides.

Additionally, the contractor anticipates completing incorporation and testing of the GALE computer code for normal reactor coolant source term for light water reactors and to the SIERRA source term module by the end of August of this year. So about the same time as the ATD module. And we're not sure yet as when we release this to the user community, we'll have both of those modules fully functional. We have to wait to see how the rest of the GALE testing goes.

And then finally, the environmental pathways and dose consequent models from NRC Dose 3 computer code will be incorporated into SIERRA in 2026. And a lot of this is progressing along the way. It is progressing because of resources and getting the resources to combine the code.

But the hope is that when we're doing this, it'll be a more efficient use of those resources we have for code development. Next slide, please. The second task that I saw we completed, which I already talked a little bit about, is improvement of the source term character -- the source term -- characterization of the source term. For normal operation phases, we actually broke this into three phases in Volume 4.

For normal operations, Phase 1 and 2 of this task, the radionuclides of interest in the normal source term include fission products, capture products, activation products produced during the normal operation of the reactor coolant system. As mentioned previously, Phase 1 will be the consolidation of the GALE codes into the SIERRA source term module. And for Phase 2, our contractor is leveraging work done by the National Reactor Innovation Center, NRIC, to implement non-design specific reactor coolant source terms for non-light water reactors from publicly available plant design information into the SIERRA source term module.

And I think my last slide in this section, I actually have kind of a breakdown of that format. So if you're curious about that, I'll go through that in a minute. And then for accident -- for Phase 3 severe accident and beyond design basis accidents, the primary source will be the source term information we gave from the work done in MELCOR and SCALE as described in Volumes 3 and 4.

And those will be put into the SNAP/RADTRAD computer code. Next slide, please. Again, this is kind of just rehashing the actual -- the inputting of the GALE code into the SIERRA source term module. Just a couple more details on that, this included adding both the pressurized and boiling water gaseous liquid effluent subroutine -- FORTRAN subroutines. There are four of them.

The subroutines have been implemented to SIERRA with some minor code changes in C Sharp and the changes to the file structure to match the existing SIERRA framework that we developed in the first task for this. The GUI and the FORTRAN based back end were decoupled for each of the future development activities and expansion of the SIERRA module for non-light water reactor technologies. Additionally, the input files one of the additional features, is now the input files for a light water reactor do not have to be present inside a GALE directory.

They can be taken from anywhere for a light water reactor. So that is making it a little bit more robust than it was before in the GALE structure. Next slide, please. Let's see. This just shows the incorporation of the GALE source term and the testing that has been done on it.

The GUI and numerical verification validation of the SIERRA source term module is underway as compared to the GALE code. And that was the GALE 2.2 I think it was. The incorporation has led to an improved user experience and will allow for streamlined development efforts moving into Phase 2 which is the non-light water reactor source term -- normal source term that we developed for the SIERRA source term module.

The GALE to SIERRA testing includes multiple facets to ensure that the user interface is functioning as expected as well as numerical testing to ensure that the calculations and the results from the modules are expected. This last slide -- next slide, please. This last slide is on the improvement of the characterization of the source is the methodology I was talking about for Phase 2. This slide on this task depicts the concepts and strategy that our contractors have mapped out for developing the normal source terms for the various non-light water reactors and fuel designs.

The proposed methodology for the normal source term will draw on the -- as I said before, the National Reactor Innovation Center fission product modeling approach and will be similar in concept to how the GALE code calculates normal source terms for light water reactors. The methodology will use built-in source term data for each non-light water reactor design and fuel design coupled with code features to determine fuel isotope concentrations, calculate the fission product release fractions to the primary coolant based upon ANSI 18.16 nuclide classes, determine activity concentrations in the primary coolant for both fission products and activation products and secondary coolant if applicable to design, determine the liquid and gaseous effluent streams for each reactor design including rates, activity, and waste stream cleanup mechanisms, i.e. holdup.

Additionally, the normal source term methodology will be flexible to allow for user defined parameters wherever because we're just putting in very generic, basic designs in Phase 2. So we think -- and as GALE was developed through the years, it was very generic and basic based on some initial calculations. And as more operating experience came about for light water reactors and the fuels, all those inputs for GALE were then improved upon in the ANSI standard. Next slide, please.

Task 3 improved the SIERRA ATD models. The third task in Volume 4 for non-light water reactor licensing siting dose assessment code readiness involves the atmospheric transport and dispersion model. Most of the licensing and siting dose assessment use or have an atomospheric transport and dispersion models which are, as I said before, typically Gaussian models. For example, ARCON, PAVAN, and XOQDOQ codes use a straight line Gaussian models with different correction factors such as building wake effects, wind direction, wind speed, atmospheric stability class, location of release points, stacked down wash, plume rise to adjust for the codes used.

The SIERRA computer code has integrated an atmospheric transport and dispersion module that has the capability of performing those same calculations for the near field, midfield, and far field calculations. Thereby the user could perform regulatory calculations relative to the three distances in the regulations. The screen capture shows an example of a setup for performing near field or the ARCON type and the midfield, PAVAN type, relative air concentration calculations used in design basis accident analyses.

The navigation inputs -- navigation and inputs from -- are similar among the legacy codes to make switching from legacy codes to SIERRA codes simpler for the users. Additionally, the decision was made during the development of the SIERRA atmospheric transport and dispersion model that the code would only use hourly meteorological data by joint frequency distributions data. The meteorological panel shown here is a simple user interface that provides the wind rows, the basic statistics about the hourly meteorological data. Also a visual summary of the meteorological data helps inform the users' interpretation of the output data.

MR. BLEY: This is Dennis Bley. Your last bullet using the hourly rather than joint frequency data with the capitals, I'm not exactly sure what joint frequency data is. But it sounds like you could be losing some correlation by the name of that data set.

MR. TOMON: Yes, and I'll show it, I think, in the next couple of slides when I show the testing. There is some differences between the legacy codes and each of the near field and midfield and the far field modeling of the SIERRA code. And some of that is based upon using hourly data device joint frequency distributions.

However, the hourly data in most cases all licensees going back to the '90s, the decision was made to use joint frequency data because the computing power back in the '90s and the early 2000s couldn't store and calculate all the hourly data that they were recording. But they have been recording that hourly data since the beginning of time. So joint frequency distributions were used to allow to do these --

(Simultaneous speaking.)

MR. TOMON: -- calculations. So actually, the hourly data is a more -- if you were to look at it in layman's -- and I'm not a meteorologist -- layman's term, hourly data is more high resolution whereas the joint frequency data is a lower resolution of the data itself.

MR. BLEY: Okay. Thank you. That helps.

MR. TOMON: Okay. Where was I? So the next --

CHAIR KIRCHNER: Let me, John, interrupt you for a moment. This is maybe a naive or foolish question. Do you benchmark this versus the MACCS modules that have ATD capability? Or is it the same capability?

MR. TOMON: It's the same capability. Right now, what we --

CHAIR KIRCHNER: That's a better answer.

MR. TOMON: Yeah, yeah. We're trying to do the -- because we're taking these outdated FORTRAN because if you really look at the PAVAN code, you have to build an input deck. You have to build FORTRAN. There's no GUI on PAVAN.

So it's really outdated and old. So what we're doing here is saying, okay, these are the standards that we're currently using for the light water fleet. This is the consolidated module. What you do with the standards what we were doing before.

So let's compare them to those legacy codes. And then from there, we will do that comparisons outside as well. But the first step is kind of getting them all into that module, and then we can actually sunset those legacy codes, those ATD legacy codes and just go on and maintain the SIERRA code with the different modules.

CHAIR KIRCHNER: Okay. That fits in with our recommendation from circa 2018 when you were last --

(Simultaneous speaking.)

MR. TOMON: Yeah, yes.

CHAIR KIRCHNER: Okay. Thank you.

MR. TOMON: So let's see. Okay. So next slide, please. Okay. So the next three slides show the results of the various testing performed on the SIERRA computer code, atmospheric and transport dispersion model. I apologize. It's kind of small. But I'll give as many details as I can in my discussion.

And if you really go one or two levels in depth, I'm going to have to go phone a friend in that regard because I'm not a meteorologist. For the midfield -- for the near field, midfield, and far field and how they compare to the legacy atmospheric transport and dispersion computer codes. That is ARCON, PAVAN, and XOQDOQ.

The testing including using hourly data from 19 sites distributed across the U.S. And those sites were picked and determined by the meteorologist in NRR. So they were actual data file sets that they're using for the current light water reactor fleet to provide varied meteorological conditions for the test cases.

Input value such as release height, stack diameter, distance to the receptor, et cetera, were varied independently within the SIERRA atmosphere transport and dispersion model and the corresponding legacy codes. Additionally, independent reviewers tested and used the interface -- excuse me, and the atmospheric modules by test cases identified in the test plan. And what I mean by that is we used a lot of our RAMP user community to test some of the -- to test the atmospheric models as well as the folks over at the meteorologist over at NRR.

This slide in particular shows the results from the testing of the ARCON code to the near field SIERRA atmospheric transport and dispersion model. When the ARCON code was compared to SIERRA for relative air concentration values, they were generally within a factor of 5 to 2. The results between ARCON and SIERRA code near field atmospheric transport dispersion model indicate regulatory consistency with the legal codes.

And the largest difference observed were for sites with high percentage of low wind speeds. And I will tell you the next -- go to the next slide, please. This is the same kind of slide for the midfield module and the SIERRA atmosphere transport dispersion module and the PAVAN code.

And the results are exactly similar, a factor of 5 to 2, good regulatory consistency. And again, where differences were observed, they were for sites with high percentages of low wind speeds. Next slide, please. And then this is the final one for the far field or the XOQDOQ comparison to the XOQDOQ code.

And again, a factor of 5 to 2 between the legacy codes. And the modules indicate -- they both -- the SIERRA code module indicates good regulatory consistency in most cases. And the largest differences again for the XOQDOQ code as compared to the SIERRA module was with high percentage of low wind speeds. Next slide, please.

This next slide shows the next steps in the computer code -- SIERRA computer code development which is to incorporate the non-light water reactor SIERRA source term, so both Phases 2 and 3 followed by work on Task 4 and 5 to incorporate improvements to the dose coefficients and environmental pathways accumulation models from NRC Dose 3 code. And as I mentioned before, we have a time line to do that. The images on this slide show the landing page for the SIERRA code with the source term module showing that you can do both the PWR and BWR and what it will look like with the advanced reactors non-LWR which is grayed out currently, as it's still under development. Next slide, please. The Task 4 will be accomplished in the development of SIERRA through the development of dose coefficients module.

Currently, the dose coefficients and dosimetry modules are hard wired into most of our legacy licensing and siting codes. And the user has a few options to edit or change them, NRC Dose 3 and SNAP/RADTRAD have hard coded values in there. And in the case of SNAP/RADTRAD, you can actually adjust and modify them.

In most of those cases, the dose coefficients are based upon the current regulations in 10 CFR Part 20. So they're based upon IRCP 2630 and 6072 models. And so theoretically, the vision would be that the dose coefficient module will be flexible enough to allow the hard wire -- to have the hard wired dose coefficients for federal guidance reports 11, 12, and 13 plus any updates to federal guidance reports, the dose coefficient such as SGR 15 and 16. Next slide, please.

The final task was the environmental pathways module. And this is going to be developed in phases. One, the first phase is the incorporation of the NRC dose environmental pathways and dose coefficients into the SIERRA code.

So that'll build out the last few modules. And we expect that to be done in 2026. We're not expecting as long a lead time as we had to get to this point because we don't have to build the consolidated framework at this point. And we don't have to build a lot of the specific modules.

And the NRC dose code has been updated in the last -- more recently in the last few years. So a lot of what we did in that to bring it in, it'd just be a straight transfer just to get it in the framework that we need. And then also in this task, we plan to leverage models from -- eventually after we get these modules built, leverage models from the Generation 2 or the GENI code, models from the decommissioning codes like RESRAD.

And then from the MAX code in Volume 3,

anything that they might learn out, we also look to maybe incorporate into this environmental pathways model. Next slide, please. This slide just shows the AGILE code design schedule for the SIERRA computer code with completed actions and the near term planning actions and milestones for the licensing and siting dose assessment code readiness within the next three years. Just to give you kind of a future, our phased approaches were almost through our near term which was when we started through the three years and starting preparations for the intermediate phase which will be the five to eight year portion.

And then with longer term being greater than eight years, and those are when we tried to -- we'll include things like decommissioning codes. Next slide, please. And then this is my final slide, and it's kind of like the summary, cut to the chase about our readiness and it probably should've been moved up. But I put it in at the end.

It shows our current status of our readiness for non-light water reactor reviews. As currently configured, the atmospheric transport and dispersion codes, as they currently exist, the legacy codes, they can do the meteorology for non-light water reactors. However, they are not very user friendly as they are currently configured as I spoke about.

Some don't have a user interface and don't have FORTRAN input decks. The goal is that the SIERRA atmospheric transport module will ease configuration issues that are experienced with the legacy code. And that'll make it much more user friendly experience moving forward, knowing that we still have -- we probably will still have more work to do in the future.

Since the SNAP/RADTRAD computer code has recently been updated to a more flexible framework, it is current ready for these reviews right now with extensive manipulation by the user. What I mean by that is there are a lot of hard built-in tables and models built into it. But the user can always user define all that information.

But that requires a lot more of the user. And it requires a lot more of the reviewer to know what the user is actually putting in. The goal in the future for this moving further is to hard wire some of those tables and those models in for non-light water reactor designs.

And we plan to build on the information in Volumes both 3 and 5 directly into those release timing and mechanisms in the SNAP/RADTRAD code. And then finally I mentioned a couple slides ago the NRC Dose 3 code was recently -- just kind of recently updated as very flexible from the standpoint that you can -- it can accept inputs -- you can accept user defined inputs from source terms or inputs from GALE for source terms. It can provide -- you can choose from the existing dose coefficients in GALE or do user defined dose coefficients.

So it is ready. But we want to add more dose coefficients, SGR 15 and 16 into that. And also we need to look at some of the other biocumulation pathways when we start talking about situations that maybe are not in the lower 48 United States, looking at those pathways because most of the models in GASPAR and LADTAP are based upon normal food consumption use of waterways and stuff that are in the lower 48. So that is something after we get NRC dose into SIERRA code -- I knew I could come up with the word -- that will then go and do any additional -- add those additional features in for areas that are more remote in those regards. And that concludes the updates to Volume 4. If you have any questions.

MEMBER BIER: Yeah, just a brief one. I assume that for most advanced reactors if the source term is small, it's going to put high emphasis on detailed fidelity within a very short distance, including even onsite buildings and things like that. Do you feel like the capability is there to do that? And again, it's just an interface question. Or are there models where you're not confident of the fidelity?

MR. TOMON: I think it's pretty much in there with the ARCON code. I mean, ARCON, a lot of the models in ARCON were adapted and brought into the MACCS code. So I think that fidelity is there already. Again, it was to make it -- to try to maintain ARCON, maintain PAVAN, maintain --

MEMBER BIER: Sure.

MR. TOMON: And resources are hard to come by. So we figured if we have one big code that does a lot, I can get resources to fix if I need them and kind of move the shells around a little bit more easily to get done what needs to get done.

MEMBER BIER: Thank you.

MEMBER MARTIN: Related to my question earlier to Keith about anticipating how people might use these codes, and certain analysis particularly for evaluating mitigated dose scenarios, invariably we'll need to propagate uncertainties. I can see the images of the GUI and everything. But you can -- behind the scenes, you can maybe run 1,000 cases in batch and vary parameters. Is there capability there?

MR. TOMON: It's not there yet. It will be built into it.

MEMBER MARTIN: Okay. That's been anticipated.

MR. TOMON: It's been anticipated. I mean, we do -- it's funny you bring it up. Probably about three, four years ago when we did our RAMP meeting, we had a non-light water reactor symposium and we got a lot of the venders come in. And they gave us a lot of direction.

That's how we decided how we were going to go with volume 4 because we listened to their feedback as well. And when we heard from them that -- right now, they run -- they'll run ARCON and for near field and they'll do it in FORTRAN and PEARL scripts so they can run 100 cases and then analyze it and do sensitivity. And they, like -- if we could do this easier, it would be better.

So we listened to them. And another thing that we found from the user community that even though SNAP/RADTRAD is designed for design basis accidents, we've had users that kind of use the flow path because it is so flexible for other things to see how the radionuclides will move around in their system and in different compartments in their system. That's what we call that component inside SNAP/RADTRAD. So we're seeing different uses for it than what it was originally designed for. And it's expanding our thoughts on what we do go forward and do with the code.

MEMBER MARTIN: I'm sure there'll be more of that. And I appreciate the comments on user feedback because that's obviously so very important from a developers perspective. And the more you communicate, the better. Time to move on?

MS. WEBBER: Yeah, I think we need just a few minutes to change out speakers for the next panel.

MEMBER MARTIN: Do we want a ten-minute break? Five-minute break. Five-minute break?

MS. WEBBER: Sounds good.

MEMBER MARTIN: So that'll just be 2:33.

(Whereupon, the above-entitled matter went off the record at 2:28 p.m. and resumed at 2:34 p.m.)

MS. WEBBER: Okay, great. Thank you.

And so, for this next portion of the presentation today you're going to hear from Lucas Kyriazidis who is going to lead the conversation with you on our fuel cycle-related activities and readiness.

And then at the end of that I'm hoping to have at least 5 minutes to provide some conclusions.

And so, with that, you're on, my friend.

MR. KYRIAZIDIS: Good afternoon, everyone. Thanks for giving us a chance to present today. So, today I'll be presenting NRC's readiness strategy for performing non-LWR fuel cycle analyses.

My name is Lucas Kyriazidis. I'm within the Office of Research. Work within the Division of Systems Analysis. Today I have my colleague Amy Bielen here and then Shawn Campbell joining myself online.

Okay. So, this slide I want to cover some of the project objectives and goals for Volume 5. The overall for Volume 5 is to ensure that we at the NRC have simulation capabilities for performing independent safety analyses for non-LWR fuel cycles.

The sub-bullets on this slide show how we'll get there.

So, we'll identify major differences between the non-LWR fuel cycle compared to the LWR fuel cycle.

We'll identify any gaps in our codes and models for performing fuel cycle analyses through exercising our codes.

We'll address any code gaps through code development activities.

And then, lastly, we'll assess, demonstrate, and document how our codes perform.

Next slide, please.

So, the approach that we took is similar to the Volume 3 approach where we first developed conceptual and as-representative-as-possible fuel cycle designs for each of the non-LWR that we analyzed under Volume 3.

So, what does a representative fuel cycle design give us? It will help identify what our codes, impact capabilities for to help improve our confidence that we're asking and answering the right types of questions. But it also helps identify the types of accidents, but along the way their boundary conditions and their boundary conditions and their initial conditions.

So, then we identify and down select key accidents to model and scale a melt core exercise and keeping online models.

Lastly, we'll develop and run simulate these representative accidents and SCALE and MELCOR to help identify where we have continued gaps or data gaps, or where we need to improve.

So, here I just highlight that, how we'll use SCALE and MELCOR.

Next slide, please.

So, this slide covers the types of analyses that we're expected to perform for the fuel cycle. This isn't an all-inclusive look, but covers the majority. So, the top graphic covers some of the accidents that we want to be able to simulate and SCALE and MELCOR for the non-LWRs.

And here we have crit safety, radionuclide decay heat generation, radiation shielding and dose, and then radiological and Non-radiological material and energy transport.

And if we dive down a bit deeper for crit safety I give an example. So, we'll be analyzing inadvertent nuclear criticality events for various fuel forms, such as solutions, powders, and even large storage arrays.

And here on the bottom of this slide I provide some of the reference documents that we used to get some insights on how to analyze these types of accidents. The NUREG/CR-6410 was a handbook that provided some insights and methodologies for performing fuel cycle analyses.

NUREG 1520 was a standard review plan for performing fuel cycle analyses.

And then NUREG 2215 and 2216 are storage and transportation NUREGs that are used in NMSS.

Next slide.

So, this slide covers the starting point of how we developed, or how we developed a non-LWR fuel cycle. The starting point was the LWR fuel cycle.

Here we took the open fuel cycle, which assumes that fuel that exits the reactor is destined for final disposal. There's no reprocessing or separations activities.

Further highlighted on this slide are the various fuel cycle stages. You have mining and milling, enrichment, fabrication, utilization, storage, and disposal. And on the legend it provides some additional details of what each of the fuel cycle stages consists of.

And then the image on the right just showed this in another format where we also talk or identify some of the regulatory areas. For example, Part 71, Part 72 storage for -- storage and transportation for spent fuel is identified. But I also heard mention of Part 20. Part 20 is also listed for dose.

Go to the next slide.

So, I talked a little bit about what the starting point for developing the non-LWR fuel cycle was and how we plan to use designs. But I also want to showcase on the slide how vastly different the non-LWRs are compared to the LWRs.

So, here on the top row I cover the baseline condition which was the LWR fuel cycle. Here I list the licensed enrichment limits, fuel forms, burn-ups, fuel residence time, whether or not we have expected fuel reprocessing, storage and transportation.

And on the following rows I present the non-LWR fuel cycles that we're looking at.

So, what I really want to highlight here is for enrichment we know we'll be looking at HALEU-level enrichment, so up to 20 percent. Various fuel forms, we've heard that we'll be looking at oxides, metals, TRISO in pebbles, compacts, and even liquid fuel. And then the burn-ups vary drastically, you can see for the two type reactors. Burn-ups are fairly low, but then for SFRs they can range up to 300 gigawatt days per metric ton yield.

And then if you hit that animation.

So, really -- well, I had some animation but that's fine.

So, really what I want to stress is if you look at the non-LWRs for storage and transportation there's a lot of TBD, which means there's a lot of unknowns that we just don't know publicly how the back end of the fuel cycle will look like. This really limits what Volume 5 will consist of and consider.

So, on the next slide, this slide covers what are the fuel cycle stages that we're considering under Volume 5. Again, the image on the left is what I've shown was the open cycle LWR fuel cycle. And the figure on the right is one of the HTGR for pebbles.

Really what I want to highlight here are just the fuel cycle stages that we're omitting or not considering under Volume 5. And on the bottom I highlight that. So, mining and milling we're not considering, power production, outside spent fuel storage and transportation, then spent fuel final disposal.

And here I want to talk a little bit about why we're omitting these things.

MEMBER BROWN: Could you go back to the other slide. And maybe this is just a quick question.

I guess I was taken aback a little bit on the storage issue. You knew a lot about what we do for storage for the current light-water reactors. You store it, let it decay, put it in casks that we know what to do with it. A single, nasty waste product that we have.

But if you look at some of these other ones, we no longer have just one really nasty waste product, you've got multiple waste products which are toxic, corrosive, can eat the hell out of everything they ever touch. And, yet, there's got to be some idea of what it takes to handle those.

And even I would ask the question why in the world are we even looking at them? But that's a personal opinion not a public opinion.

But it seems TBD doesn't get factored in terms of an assessment in this overall fuel cycle process. It seems to me that it ought not be ignored as opposed to -- because it's probably the worst of everything, particularly the ones where you mix the fuel in with the coolant, which is really tasty.

MR. KYRIAZIDIS: So, that's a, that's a good, good point.

When I say we're not considering it under Volume 5, we're not considering it initially until more information becomes available. So, you can treat

Volume 5 as maybe like an iterative process. As more information becomes available we'll go back and reassess our codes to say whether or not we have the capabilities to model it, or if we need to add models or perform sensitivity studies.

But to go back to your point of why this is all listed TBD, we can leverage historic information. How EBR-II stored their waste, how pebbles were stored. We know how they're stored on site, we just don't know past that stage how they will be stored for long-term storage.

MEMBER BROWN: Does anybody ever look back at the original *Sea Wolf*, the submarine sodium plant, and how that one was handled? I mean, there's an historical perspective. It's not classified anymore, it's public. The prototype and submarine was built and you sure as heck -- and that was a sodium reactor. It's lifetime was very short because they couldn't keep it from waking and causing other problems, and freezing all the time.

But all I'm saying is sodium isn't in many of these, however form you look at it. It's just whether you need more information on most of, a lot of these things, they've really got to be categorized as totally -- even though you have limited experience, if you look back at these other ones they're limited experience, it's not like they've been trying to be used for decades, because you can't.

MEMBER PETTI: You're aware of how they dealt with this on the submarine. The question is that how applicees will deal with it.

MEMBER BROWN: Yeah. Well, thank you. That's a -- I'm sorry, just had to get that thought in.

MS. WEBBER: Those are good comments.

I also think that, you know, we didn't get started on this particular effort until more recently. The research that we've been doing has been underway for maybe 3 years because, you know, the focus was initially on the first set of the volumes.

But then as we started to interface with the Office of Nuclear Material, Safety, and Safeguards, Division of Fuel Management, we started to realize that we really need to take a look at how our codes can be used for any one of these fuel cycle stages.

So, so what you're saying here is maybe less progress relative to the other areas that we focus. So, you're pointing out, you know, a really good source of information in both cases. So, and that we'll have to continue to take a look at that.

MEMBER BROWN: Okay.

MR. ESMAILI: Can I add that, you know, we don't want to be held up in these fuels, right. So, one of the things that we are doing is just this is on code, you know, code readiness, right. So, as the, as Lucas mentioned, you know, as information becomes available and this is our code readiness then, then we can, you know, we have the flexibility to do this.

At this point, since we do not know, we don't want to, you know, expend our limited resources on things that we do not know.

MR. KYRIAZIDIS: Yeah. And that's a good point, too. We do have limited resources. So, rather than propose what-ifs could happen for the back end, we focus our resources on areas we have confidence in.

So, if we propose a back end to the fuel cycle and we're completely wrong, well, then we have to go back and re-do those efforts. So, rather than -- and that focused towards one of the -- one end of the fuel cycle and some fuel handling accidents. And then we wait for the back end. As more time progresses, more information becomes publicly available, then we'll start looking at that back end.

We can go to the next slide.

Okay. So, on this slide I present one of the first deliverables we issued under Volume 5. The image on the left is the report that documents all of the fuel cycle designs that we came up with.

There's a design for, I mentioned there's a design for each of the non-LWRs that we looked at. So, there's a fuel cycle design for the heat pipe reactor, FHR, HTGR, SFR, and MSR.

We issued this in December of 2023. And it covers UF6 enrichment all the way through onsite spent fuel storage and transportation.

Let's see if it's here. Okay.

So, I want to dive into one example, the MSR fuel cycle. And I'll talk a little bit about the various fuel cycles they used and some of the highlights that the report mentioned.

So, we looked at UF6 enrichment. And that really dives into process of conversion and the gas center fusion, and identifies what are the hazards, the chemical hazards and also radiological hazards, associated with that fuel cycle stage.

Then we progressed to transportation of UF6. In this we identified a potential transportation package that could be used to move 20 weight percent UF6. We assumed it was the DN30-X. I can say now that that is an NRC-licensed package now.

Then we progress. We go to fuel salt synthesis, and we dive into several examples. We look at both thermal spectrum systems, but also fast spectrum systems.

We go into the chemistry of how fluoride-based salts are produced for thermal systems, and then fluoride-based salts for fast spectrum systems.

We identified the steps needed to do salt synthesis, but also some of the chemical hazards.

And then we look at the salt, salt transportation, where it will happen. You've got your carrier salt, you've got your fissile salt. Will it happen on site?

So that kind of dives into U1, and then it will go all the way to onsite waste treatment. But for the sake of time I won't go through those. You can go (audio interference).

So, now I want to cover some of the accidents that we looked at under Volume 5. I only intend to cover three, but we've looked at quite a bit to date.

Ranges from HT -- well, we've gone through our chops on the ACGR and the sodium fast reactors, and we've modeled several accidents for the various stages of those fuel cycles. But I'll cover three today.

I'll cover water ingress during transportation of the UF6 shipping package.

I'll cover UF6 cylinder rupture within the fuel facility. That was a result of overfilling and heating the tanks.

And then, lastly, we'll look at a more complicated accident where we dropped the spent fuel assembly of an SFR type in the containment building.

And we'll look at doses and we'll look at the material transport throughout the building.

So, I want to -- I'll just glance on this. This is the UF6 enrichment fuel cycle stage. And here I just list some of the hazardous material that we identified in the research. One note is UF6 is the only -- was the only radiological hazard associated with this fuel cycle stage. And then some of the potential accidents that we identified.

So, the first accident I want to talk about is the UF6 cylinder rupture. Here we are assuming a 48Y cylinder is overfilled, heated, and eventually goes through a catastrophic failure, essentially emptying out all its inventory within the storage compartment.

Here we're using MELCOR to model the radiological transportation of the material, but also to gauge how much vapor and aerosols are released.

The image on the right is the MELCOR model that we put together. Here you can see, what's important is you can see the UF6 storage area, where it is in the storage compartment area, all the intakes and exhaust and the doorways, so essentially where the material can be transported. And then, eventually, how it's connected to the environment.

What's important here is that it's going to go through a building filter, so I do want to highlight that.

I've also put some of the modeling assumptions that we made. We assumed 14,000 kilograms was loaded or is emptied out of the container. And that is an instantaneous release.

Then, lastly, I do highlight this chemical reaction because UF6 does interact with water. So, you're going to be producing the UO2F2 and then hydrofluoric acid. And MELCOR will also track or estimate some of those findings.

Go to the next slide.

So, here I present some of the results that we showcased at the HTGR fuel cycle workshop.

So, here on the top left is material species. So, what we're showing here is that instantaneous or the chemical reaction where we're assuming UF6, interacting it with water to produce hydrofluoric acid.

You can see that because it was an instantaneous release there's a jump followed by a slow increase in hydrofluoric acid. That's just showing that as it interacts with the air and the water it's continually being formed.

The bottom set of figures show the transportation of UF6, UO2F2, and hydrofluoric acid. What's important here is to look foremost to say whether it's in vapor or in aerosol. And that is important because of the release mechanisms for both of these. If it's an aerosol it should be expected to get picked up through building filtration. If it's a vapor -- and then, you can go on.

So, that was it on the material, the cylinder rupture. I do want to maybe dive over there and to have a little bit more time to go through the crit analysis for the DN30-X.

So, here the DN30-X was a UF6 transportation package that's built with neutron poisons or constructed with neutron poisons. Here the image on the right shows that. Here you can see the control rods, the UF6, and then the PST. Essentially it's a shell-in-shell shipping package where you have an outer and inner metal container.

So, here we looked at the reconfigurations. We looked at an infinite array of these surrounded by air.

We looked at a hexagonal array surrounded by water. There's no water ingress between the PSDs, both the outer and inner shells.

And then the third array, or the third configuration is we looked at an infinite array surrounded by water, with water ingress in between the outer and inner shells. Here's we're using SCALE shift to perform a crit analysis. We're using both ENDF VII.1 and ENDF VIII so we can gauge the difference between the Nuclear Data Libraries.

And then on here I say that ship this SCALE's new Monte Carlo high performance neutron transport code. And then also some of the assumptions.

All the assumptions made here were bad conservatives, conservatisms to the models. So, by neglecting the thermal insulating foam we're promoting neutron communication between the shipping packages.

And then we're using an elevated density to increase the solid material. And we're overfilling these shipping packages.

Here we assume HF as in curies, so we're adding some moderation, too, to the system.

So, really what we're trying to showcase here is that SCALE can used to model HALEU-level enrichments, making sure that there's no gaps in the codes. And then what we're seeing for the results confirm what we would expect to be seen.

So, here I present the results of the three cases. The image on the left is the cross-sectional cut of the 10 weight percent cannister. So, we looked at two, two configurations, the 10 weight percent and the 20 weight percent just to compare them.

The table on the left shows our baseline condition. So, this is the infinite array in care. And here you can see that the shipping packages were substantially subcritical. The differences between the ENDF libraries were minimum, too. So, they were consistent with each other.

The middle figure shows the second case where we have water surrounding the outer of the package. And here we also added a new variant where we're adjusting the spacing between the shipping package. But X equals 0 on this figure. These shipping packages are essentially touching on the outer shells.

And then here what we're showing is by introducing water we're actually decreasing K effective or the reactivity of the system. Here we're adding moderation, so it's an over moderated system, so we're increasing the parasitic capture of the water.

And that's also shown by increasing the distance between the shipping packages you're increasing the amount of water in the system, which is why you're seeing that decrease in K effective.

And then the third figure where we have water ingress between the inner and outer PSDs, it's just also increasing the water in the system, over moderating the system, and you're increasing neutron parasitic capture.

We can go to the next slide.

Okay. So, the last action I want to cover, which is the most complicated action that we looked at, it's during refueling or spent fuel operations for the SFR.

So, here we'll look at the doses estimate, or the dose analysis first.

So, during refueling operations the refueling machine that's used moves fresh and spent fuel out of the reactor and into the storage rack. So, here what we're assuming is that during unloading operations the spent fuel assembly undergoes failure during a seismic event or a ring failure. The spent fuel assembly is dropped within the containment building. And now we want to look at what's the dosage inside the containment building and outside the containment building.

So, here we're going to be using SCALE to develop the irradiated source term, but then also perform the 3-D shielding analysis. And here on the image is just what the ABCR building looks like. It's a little blurry, but you can see where the spent fuel refueling machine is, the unloading machine, where the cask is.

And on the next slide I will show where the spent fuel assembly is dropped.

So, we looked at two cases here, two types of fuel. We looked at just U, the binary metallic fuel U zirc. Then we also looked at a uranium plutonium fuel. We call that, the U zirc fuel is HALEU fuel and then the U tru, or transuranic, is the uranium plutonium fuel. We looked at two cases.

We also looked at two different cooling times. What's shown on this figure is the more extreme case where you can think of it as two accidents. This did not have sufficient coolant. Typically, spent fuel will be held maybe for seven cycles of 28 months. Here we assumed 10 days of cooling, so it was an inadvertent picked up spent fuel assembly that was going to be loaded.

So, here what we are showing are the doses inside the containment building. Here the spent fuel assembly is dropped against the containment building. That arrow is pointing right where that peak dose is. And then the figure on the right that shows the dose outside of the building.

So, here what we're showing is that, yes, SCALE can be used to generate your source term for various types of fuels. Here we looked at SFR fuel. And I think there was a question during the max whether or not you can look at activated steels or activated sodium. And this source term we did assume that. We were able to, we did account for sodium activation, we did account for stainless steel activation. That made its way or was accounted for in our irradiated source term.

And then not highlighted here, but in the workshop material we did look at assume this is a PWR extended fuel assembly, are the, are the comparisons somewhat in the same area? And, yes, the SFR fuel was about four times higher. But we all got in the same order of magnitude if we were to drop a PWR spent fuel assembly on the containment building floor.

You can go to the next slide.

Okay. So, now we're going to look at another material transport back here using MELCOR.

Here it's a similar accident where we're taking the spent fuel assembly, an SFR type, we're loading it into the inter-building cask, that's going to be essentially destined for onsite spent fuel storage. But the crane fails, drops this cask. And now we want to see what happens to the fuel.

What happens, we fail the fuel assembly, then if we do does the material -- where does the material end up?

So, here some modeling assumptions. There's no active cooling. All the active cooling is assumed to have failed as well. There's no residual sodium inside the cask, so you just have a loaded bare assembly inside of a shielded cask. We want to see what happens to fuel temperatures and then where that goes.

So, the figure on the left shows several sensitivity studies. So, we looked at another actual case of essentially loading a wrong assembly. After one day of shutdown we loaded a spent fuel assembly into the cask. The cask fails. There's no active cooling on the cask. What happens to the fuel?

This figure on the left shows, that blue line shows what happens. So, after about 40 minutes the fuel assembly fails. You've hit your T-clad or your cladding limits and melted.

If you picked up the right fuel assembly after seven cycles, you can see those other line graphs, you maintain your cladding integrity. You don't fail the fuel.

So, we just wanted to show some capabilities of what ifs.

The figure in the middle just shows the K-eff as a function of time. You can see that the one day cooled spent fuel assembly was several orders of magnitude higher.

And then, lastly, the figure on the right just shows, okay, you failed the fuel assembly. Where does the material go?

Here we're just showing containment and

settled radionuclides. The workshop also showed some environmental releases. But for this, for the safety I didn't present that.

So, so now I'm going to transition to where all these deliverables, and workshop slides, and videos can be found.

So, on this slide it is showcased. The publicly -- the public webpage where we can, where we store all this material. So, the QR code will take you there.

So, today we've done a fuel cycle analysis for the high temperature gas cooled reactor, one for the sodium fast reactor fuel cycle. And our next planned workshop is for this summer for the MSR, the molten salt, molten salt fuel reactors.

Then here I list some of the accidents that we plan to cover. We'll look at some crit analysis during fuel cell conditioning.

We'll look at some beryllium releases.

And then we'll also look at estimated doses on the primary heat exchanger.

So, key conclusions and highlights.

So, we have revealed some information gaps in our work to date. We have noticed that there is no commercially sized transportation packages for moving fresh pebbles.

There's a lack of information on how these are going to be stored long-term and even onsite. We had to make some assumptions here as well.

But we don't envision these challenging our codes. We think of these are mere, like, geometry changes. If we're doing a different shipping package they may be bigger, they may use some different material. But we don't think that we're missing anything fundamental in both SCALE and MELCOR.

We have noticed that we do need validation data, specifically crit safety benchmarking data, especially for uranium graphite-based systems.

Here I do want to note a new collaboration between NRC and DOE that we're working to fill this data gap. It's called the D&CSH program, or the Development and Criticality Safety Benchmarks for HALEU fuel cycle and transportation. Here, essentially, this is, the goal here is to produce high quality, publicly available benchmark data, nuclear data, and evaluations for a wide range of HALEU systems.

This was enacted under the Inflation Reduction Act, I believe. So, this is underway. We had our first workshop in February of 2024. So, the goal here is just to produce or fill a data gap that's being noted throughout, for emerging also industry.

MEMBER PETTI: On the TRISO stuff, you're worried about benchmarking like in storage configuration?

MR. KYRIAZIDIS: Yes. Yes, yeah.

MEMBER PETTI: Okay.

MR. KYRIAZIDIS: Criticality basically.

MEMBER PETTI: Because there was reactor benchmarks done.

MR. KYRIAZIDIS: Yes.

MEMBER PETTI: But I would wonder if you could contact KFA in Germany. They decommissioned the AVIA, AVIA NTH, yeah. And they, you know, they have casks that handle the pebbles. Their regulators must ask some questions.

Maybe there's some data there. That's something that I could think of.

MR. KYRIAZIDIS: Yeah. That's a good point. I noted that. And we can see if we can.

So, we've demonstrated some of the accidents. We are ready to support fuel cycle analyses. We have licensed the UF6 shipping package for the BN30-X.

There are other shipping packages that

have been licensed to move fresh fuel or fresh pebbles. NMSS has approved two that can move compacts and pebbles. So, we are, have used the SCALE to perform some of these analyses.

And I want to leave you with some next steps.

We do have some co-development activities underway. We're adding some flexibilities in the geometries that we can handle with SCALE. The last two points cover that.

We're looking to add some controlled blade, being able to model controlled blades with the pebble system, being able to handle complex arbitrary geometries. That's easy to model. That really would be used to model fractured pebbles.

Or, if reprocessing does ever come, looking at metallic fuel finds and potentially taking the burden off the user trying to model these complex geometries and having SCALE do that.

And then also some MELCOR improvements where we were looking at adding multiple working fluids and origin integration into MELCOR or MSR analyses.

And then I want to touch upon maintaining awareness of industry priorities. Earlier in my talk

I touched upon a lot of the unknowns for the back end of the fuel cycle. And this is what I want to stress is that we are maintaining awareness of what, what is being proposed, what technologies are being proposed. And then, continually assessing do we need to do anything to our codes to be able to model this? If so, make it happen. If not, we can address it through some sensitivity studies.

And then, lastly, training and knowledge management.

We'll continue to hold public workshops to highlight our capabilities, but we'll also hold some internal staff training to pass on this expertise so we can use these codes.

MEMBER PETTI: Once again on TRISO. It is a commercial shipper that is going to be shipping fresh fuel. And I would imagine NMSS gets involved in that.

MR. KYRIAZIDIS: Yes.

MEMBER PETTI: Right?

MR. KYRIAZIDIS: Yes. Yeah.

So, we --

MEMBER PETTI: Data was developed, very recent data was developed to support that that you guys should get access to.

MR. KYRIAZIDIS: Yeah. I should say that the two shipping packages that were approved were fairly limited in size. And so, we do want to ask a question on SCALE I know.

MEMBER PETTI: This one I'm not sure it's fully approved yet.

MR. KYRIAZIDIS: Oh. Oh, oh.

MEMBER PETTI: And it's going to be significant. It's going to be pro loads, a pro load. You know, not all in one.

MR. KYRIAZIDIS: Yeah.

MEMBER PETTI: But multiple big, big shipments.

The ones, I think the ones you're talking about are the one that --

MR. KYRIAZIDIS: Shipped invert.

MR. KYRIAZIDIS: Yeah. Yeah. That's, no, there's a much bigger effort underway. And I can't speak to it. But if it's commercial it means it's going through NMSS.

MR. KYRIAZIDIS: Yeah.

MEMBER PETTI: The data is going to be somewhere at NRC. And you guys ought to get access to it.

MR. KYRIAZIDIS: I think that ends my talk.

If there's questions, glad to take them.

MEMBER MARTIN: Any questions from the members, consultants?

Hearing none, let's go wrap up.

MS. WEBBER: Okay. Thank you.

Okay, so this has been a really good meeting. I really appreciate the candor, the comments, the questions, the references of additional information.

You know, when I, when I think about the history of licensing large light-water reactors, and co-development capability that's been developed, you know it was developed over decades, 40-plus years. You know, we have very mature processes when it comes to modeling and simulation.

And while it's, you know, it's nice to and it's important actually to take a lot of those, maybe what I would call standards of code development activity, and bring them forth into this context, you know, it's a little bit different context. We have large light-water reactors, we have pressurized systems, you know. And we've studied those reactors for quite a long time.

Here in this context of non-light water reactors it's a new, it's a new world really. With fewer pressurized type reactor designs, we still have a few that are pressurized. And, you know, for us the challenge is then, you know, priorities.

There's not really been too much down selection in terms of funding. Although we've been following, you know, DOE's funding stream through the advanced reactor development or demonstration project, and all of their other funding programs to figure out where we place our very limited resources.

And so, this program that we've developed has had fits and starts over many decades. Back in NGNP days there was some code development that was done in some context. But more recently since we started this initiative, you know we're expanding our view to look at all of the capabilities that we have in our codes.

And I think, you know, where we are today is obviously not where we are with the light-water reactor capabilities. But I think we have made significant progress with the funding that we have had, which is not -- it's definitely nowhere near the funding levels that other organizations have for their modeling and simulation capabilities.

And so, you know, in my estimation, you know, we have generic capability to evaluate more simple situations, and then we're developing capabilities that look at more complex design aspects. As Steve talked about, we're doing CRAB.

But, but I believe that, you know, we've done a, in my, you know, my opinion, my staff contractors and close collaborations with the labs, and even international organizations have done what we can with the information that we have available to date. Recognizing there's not a lot of operating experience with these non-light water reactors. There's more in some reactor types, like sodium fast reactors, or maybe even high temperature gas reactors. But then in other areas there's a lot less data, a lot less operating experience.

So, if we could go to the next slide, please.

I really have to thank you so much, you know, for your questions, your comments, references to other information. I think that goes a long way to helping us with the progress that we're trying to make.

I also hope that you've seen from the presentations today that we've made significant progress with our codes capability and developing staff expertise.

On the left-hand side of the slide I note a few activities that we have completed to date. And I won't read them, for the sake of time. But I also want to note that, you know, we still have more, more work that can be done. And your questions have elucidated, you know, some areas that over time and with budgets we should take a closer look at.

I think with any kind of model or simulations program it's an evolution. And it depends on what the problems are that you're faced with that you're trying to evaluate. I think, you know, from a regulatory standpoint, you know, the agency's taken a perspective of conservatisms. So, where we have uncertainties that are fairly large, you're building conservatisms relative to our safety findings. And we use regulatory tools to address some of those shortcomings.

So, all that's to say that, you know, in the area that we're focused on, which is modeling and simulation, you know, we still plan to update our reference plant model to be able to add additional capability as more information becomes available.

We plan to continue our verification and validation efforts, update codes with new models, continue the code consolidation effort in the siting and licensing dose assessment area. And we'll continue to hold public, public meetings and demonstration workshops.

And, you know, the one thing that's been said a couple times through the day is that the staff is building their expertise by learning and doing. So, while they're working on the code building activities, running the codes, they're building their expertise. And that's going to be really important to support the licensing activities.

You know, admittedly, one challenge that we're facing now is as we continue our code development efforts we're starting to experience fairly significant budget reductions. And that's going to put us in a position to have to really very deliberately focus on how we place those resources. And we may not be able to, as we've done in the past, be able to fund all of these different areas that you've heard about today. If we face budget shortfalls we may not be able to do that.

So, I can't stress enough how important it is to have a letter from you to identify, you know, what you think about our progress areas, where you think in the near term, given the environment that we have here today, you know, where our focus, time, and resources could be spent, I think that's going to be very helpful to our program here.

So, I really want to thank you again for your active participation. It's been a fruitful meeting for our staff and I. And I look to continue dialog as we move forward in this area.

And with that, I will open it up for any questions or comments that you have.

MEMBER MARTIN: Well, I'll go start, you know, thanking you and your staff. I think I can speak for the committee that what we've seen here is a remarkable accomplishment over a broad scope of work.

The subject of, you know, code development, deterministic safety evaluations, you know, on the surface just sounds like, you know, very researchy and, you know, we're trying to match data. But in the context of safety we look at the evaluation model concept and the ability to make decisions. That's what we're ultimately trying to do.

You know, applicants will bring in a rock and we'll need to review it. First we have the time pressure to do it quickly, to do it with confidence. The emphasis on D&B is, of course, tremendous to present evidence that the tools that we are going to be checking and, of course, in some cases it may be the same tools as the applicants, you know, will be using. But, you know, we have to have general confidence.

The reference models obviously help you develop staff. That competency is invaluable. It also will help you to move quickly into developing models.

What I heard was, you know, your engagements with the stakeholders and partners like, of course, DOE, the user community, to some extent international. I certainly would encourage expanding those communication channels.

I think my last thought is, of course, you know, thinking about how people will be performing, you know, non-light water reactor safety analysis in a Part 53 or NEI-18 forum where they're answering questions about how good the design is, or safety classification of SSCs, defensive, cliff edge effects.

That's only beginning to scratch the surface on, you know, how the agency and people will expect safety analysis to look like. And I think it's a lot of fertile areas moving forward.

So, I'm excited for what I've seen and certainly the future that is out there for, you know, research to contribute and answer questions that will need that evidence to help make decisions.

So, and I'll turn to my colleagues. Do you have any other comments?

VICE CHAIR HALNON: I just have one question.

Kim, it's sort of, I don't know if shocking is the right word, astounds me that if people understand the function to do the budgeting, if they understand the importance of this code development in the springing of the advanced reactor world, since we don't have operating experience, it astounds me that you're feeling stressed on budget. I don't mean a blank check.

But certainly so that obviously needs to be a key point that we need to make is that the enabler has to be the expertise and the getting and gathering of data, and then the disabler can be the budget.

MS. WEBBER: Correct.

MEMBER BIER: I guess I have a quick follow-up on that, which is it your sense that the budget constraints are due to just kind of across-the-board cuts within the agency or priorities of other things that are getting increased funding?

MS. WEBBER: You know, it's really hard to say, to be honest with you. You know, there's perspectives across the agency at senior levels about, you know, doing the right amount of work. And so what all of the agency's trying to figure out what does that look like.

So, this is part of it. You know, this is part of the agency trying to figure out what's the right level of effort. What's the right amount of work to do what we do -- need to do to make the right safety findings? And so, we're just caught up in that.

MEMBER BALLINGER: I have a couple of sort of high level questions.

We heard an awful lot about the code development efforts that are necessary. I'm assuming that you've prioritized based on the plants that you anticipate to come in.

MS. WEBBER: Yes.

MEMBER BALLINGER: So, my first question is what's the long pole in the tent for that? You've got all this stuff that you're doing. If all of a sudden they show up at the door and say, here's the submittal, what is the highest priority of all of this that you have to do?

And the second part of that -- it's a two-part question -- is that do you have enough information now to ballpark the results?

You know what I mean by ballparking?

MS. WEBBER: I'm not quite sure of that context.

MEMBER BALLINGER: You simplify things to the extent that you can almost do a back of an envelope calculation to find out if you're in deep yogurt.

MS. WEBBER: Okay.

MEMBER BALLINGER: Right? Well, so that --

MEMBER BROWN: There are other frames of reference here.

MEMBER BALLINGER: -- once you've identified those things you can then also apply that knowledge to the focus that you need to have. So, it's more of an organizational kind of thing, given the budget constraints that you have.

So, have you done that analysis so you know what the long poles are? You know whose submittal is coming down the pike.

MS. WEBBER: I think the way -- so, let me start maybe on the answer and then I'll look to my other folks to chime in.

But, you know, we know that with the advanced reactor thermal project, we know that those are fairly certain --

MEMBER BALLINGER: Yes.

MS. WEBBER: -- given the funding levels. And the application Kairos Power submitted, you know, is here.

So, we've been following that the last several weeks to make sure that our capabilities are available to support those two big type reactors.

You know, we've also been watching, you know, all the changes that have been going on with the heat pipe reactor community. And, you know, we de-emphasized that. But, you know, you never know who's going to submit what and when because there's a lot of money out there. And so, while we are following the signs of where the funding is going, you know, we are doing some work in those other areas, as you've heard from the staff.

So, you know, it's hard for me to say what the long pole in the tent is because we're faced with trying to figure out what's right in front of us in terms of being able to support those analyses.

And I would also say that, you know, if we -- you know, we're trying to build the capability to address some of the comments made here today. You know, to be able to have simplistic analysis capability, you know, like we used for the Hermes construction permit application. It was a very broad brush kind of analysis but it gave insights for the level of safety findings and effort that was needed at that time.

Now, when you get into sort of an operating license stage where there's going to be a need for additional analysis, you know, we're trying to anticipate what that looks like. And that's where maybe we need some of the more detailed efforts in our, for example, in our CRAB, through CRAB codes.

So, you know, we're not quite sure what we're going to get, but we're trying to make sure that we have the capability to fit as much as we possibly can.

So, I don't know, I hope that partially --

CHAIR KIRCHNER: I'm going to have to take my duties as chairman and interrupt because we are up against a time limit here.

We still need to allow the public to make comments. So, let me interject myself here and say to those who are participating online or in the -- with us here in person, if there is anyone from the public who would like to make a comment, please state your name and affiliation, as appropriate, and make your comment.

Hearing no one volunteering a comment, we're getting a lot of background noise because we have people who are joining the open line for the next scheduled meeting.

Bob, I would think I have a different vantage point than you do simply because we saw this at its inception back in the 2016, '17, '18 time frame. And I would just observe that they've made significant progress across a broad array of technologies that each have their rather unique challenges and requirements in terms of modeling and simulation capability.

I don't know that we can weigh in directly on budget, but we can and we have pointed out to the Commission that with these new concepts they are not going to come with a lot of operating data and so on. So, the emphasis on modeling and simulation, and its importance as part of the licensing case and establishing the safety case for these concepts is extremely important.

So, with that, we are going to look to you to --

MEMBER BALLINGER: Sure.

CHAIR KIRCHNER: -- write out a draft letter and socialize that with the, with the members. And we'll take this up again.

MEMBER MARTIN: I feel like I got what I need to kind of get started --

CHAIR KIRCHNER: Okay.

MEMBER MARTIN: -- on that.

Again, as you've noted, there was a lot here. And, of course there's a lot in the material provided at the meeting.

And I'll draw from that and have that ready here certainly before our next meeting.

CHAIR KIRCHNER: Okay. So, with that we, for those of you joining us we were scheduled to start at 3:30 picking up the NuScale SDAA chapter reviews.

We'll take just a short break to change gears here, and try and start at about 3:35 p.m.

I apologize for the delay.

MS. WEBBER: Thank you very much.

CHAIR KIRCHNER: We'll temporarily recess then.

Thank you all. Thank you, Kim, and all the presenters.

(Whereupon, the above-entitled matter went off the record at 3:28 p.m. and resumed at 3:37 p.m.)

CHAIR KIRCHNER: Okay. We are ready to begin.

And we are turning now to our review of the NuScale Standard Design Approval Application, and today, we're going to take up members' assessments on Chapters 2 and 11 and 17. And with that -- oh, excuse me -- I left out 13 as well.

So, we're going to start with Greg Halnon who's going to go through his assessment on Chapter 2, and then, we'll continue with Greg on Chapter 13, and then, loop back and do the rest in order.

So, with that, Greg.

MEMBER HALNON: Thanks, Walt.

I'm just going to talk in an overview at this point. Okay?

CHAIR KIRCHNER: Yes, I think so.

MEMBER HALNON: So, Chapter 2 is, understandably, for a standard design application, would be sparse because there's no site characteristics. However, there are some items and they develop a set of parameters that set the boundaries for site selection and construction.

It's sort of like a plant parameter envelope-type thing. I don't want to call it that or put a label on it like that. However, there are some parameters.

And rather than go through a lot of detail and all the different parameters and stuff, because it's pretty well done in both the SER and the SAR, the only one that stuck out to me was the fact that the site for the precipitation studies that were used to develop the parameter about -- I'm going to roughly say so many inches -- 19 inches, or whatever, precipitation. Or I can't remember what the exact number was.

But that HMR study -- and we went through this a lot with the Fukushima flooding -- is storms, going all the way back to the 1800s and coming forward, the way they collected that data was they would go out, after a big storm, they would go out to a farm and find a coffee bucket that was, a coffee can that was full of water, and they would measure it and figure out how long it -- it was not super scientific, but it was hard data. Nevertheless, it's served us well and it is, as the staff suggested, very conservative in many ways.

But when you transposition a study of a storm, you know, you make a lot of assumptions on topography and mountains, humidity, and all kinds of currents, and whatnot. And obviously, our meteorological capabilities since the 1970s, when this report was written, are much better now with the radars and whatnot.

And given that, along with the scientific evidence and other people's opinion on climate change, in the recent law that was passed -- the infrastructure law I think it was -- NOAA has been given quite a bit of money to redo the precipitation studies, not specifically just that. I mean, they're doing a lot of studies, but one of them is the precipitation study, which makes sense that future applicants are going to use the most recent studies to do that.

And the way it's set up is that an applicant can come and say as long as I'm within these boundaries, I can put it on this site. So, that's based on an old study.

My point during the Subcommittee was: should it be better to base that on the most contemporary study? And, in fact, the staff responded back -- and I think if they want to enlighten us a little bit more on it, they can -- that, in a way, because when they do the flooding study, the Reg Guide requirements for flooding studies, not precipitation, but flooding, requires the use of the most contemporary data. So, when study gets done, the sites will use the contemporary data. In the meantime, the HMR-52 study is conservative and has served us well.

So, given the response that the staff came back with and the roadmap they showed how the flooding study would pick up the most recent data for precipitation, I'm okay with that roadmap. What I would like to do is revise the memo to show what that roadmap is in a very high level, to show how that issue is picked up as we go forward.

I think it's an important issue. I think it's not just the scientific and regulatory issue, but this could be a political issue as well. And if you're using an old study and someone comes into a hearing, what are you going to say when, you know, they based this site on a 1970-something report that had 100-year-old studies in it? And we have this brand-new one over here that shows something different. It could be an interesting discussion during a hearing for a site, siting of a reactor.

So, to me, it's important to do that. So, I do need to revise this memo, because this memo has left an open question. And I feel the question is adequately responded to.

CHAIR KIRCHNER: Were there any other points you wanted to make at this point? From a process standpoint, since you're the first one up, I'm coming back at --

MEMBER HALNON: Breaking new ground.

CHAIR KIRCHNER: -- how we did this the first time around, and then, made, also, a more recent application.

We have the individual memos from our colleagues. We, basically, have read then into the record.

MEMBER HALNON: Yes, but I think that what they don't have in the record is a response to the question. And rather than me characterize it, I would like the staff to characterize it.

CHAIR KIRCHNER: Yes. So, we do have staff here to address that.

MEMBER HALNON: Right.

MEMBER PETTI: Your revised letter would capture that?

MEMBER HALNON: Yes, but I think it should be presented in a public forum.

MEMBER PETTI: Oh, okay. Is that what you said?

MEMBER HALNON: Yes.

MR. SNODDERLY: Yes, and this is Mike Snodderly, Senior Staff Engineer for the ACRS.

I have the staff's response to the two issues, the one for Chapter 2 and the one associated with Chapter 13. And what I would like to propose is that, for the benefit of those listening in from the public that haven't seen this new information, I would read it in. And, of course, both of these responses will be included in the transcript.

And then, yes, based on this new information, I would recommend that Member Halnon revise the memo, and then, we can take up Chapter 2 tomorrow, and then, we can figure out what to do with Chapter 13. And then, I expect the other three memos for 10, 11, and 17 are a little more cut-and-dry and ready to go.

But, eventually, I think being consistent with what we did with Kairos, we would go line-by-line, so that members have an opportunity to give feedback to the reviewing member, to kind of say, yeah, we're all onboard with the recommendation for either no further review or we need further review in these areas.

So, if that's okay, the first response is just a half-page, and I'll read it into the record.

So concerning the issue of hydrometeorological reports, the staff does not consider it to be necessary that the NuScale standard design approval application include a statement requiring a site-specific precipitation study with the use of the most contemporary NOAA HMR report -- HMR stands for hydrometeorological report -- or equivalent.

To ensure climate change is accounted for in the meteorological sections impacting the design, SDAA COL Item 2.0-1 directs future applicants referencing the NuScale US460 design to demonstrate that the site-specific characteristics are bounded by the site parameters specified in SDAA Table 2.0-1. If those values are not bounded, then the applicant will demonstrate the acceptability of the site-specific values.

If new precipitation studies are available at the time of the application, then the applicant should follow the guidance provided in Draft Guidance 1290, soon to be Revision 3, of Reg Guide 1.59, which states that, `The probable maximum precipitation values provided by the HMRs should be evaluated in light of precipitation events that have occurred in the region since the HMRs were published.' If an alternative source other than an HMR prepared by the National Weather Service is used for the PMP estimate, the basis for the specific PMP value used needs to be explained.

Considerations on an acceptable approach to the estimation of a site-specific PMP as an alternative to an HMR-based estimate can be found in NUREG/KM-0015.

Current NOAA HMRs provide conservative extreme precipitation estimates and are accepted by both the NRC and the nuclear industry. When new data from NOAA or the National Academy of Sciences is available, the NRC will review the data and update the guidance, as appropriate.

Any applicant referencing the NuScale US460 design must demonstrate that the site is able to be protected against extreme precipitation and is bounded by the site parameters identified in SDAA Table 2.0-1.

And as I said, that response will be included as part of the transcript when it will be made available.

MEMBER HALNON: Thank you, Mike.

Just for reference, that Reg Guide 1.59 is the Flooding Evaluation for Sites.

CHAIR KIRCHNER: So, then, I think, Greg, we should just go to your conclusion and recommendation for the record on Chapter 2, unless there's need for further elaboration.

MEMBER HALNON: Yes, well, the recommendation was to consider using, basically, what they said, to do a site-specific study for each use of the SDAA. And, of course, that's not going to be necessary because they will be doing Reg Guide 1.59, Rev. 3.

CHAIR KIRCHNER: Right.

MEMBER HALNON: The only thing that's kind of an open question is, when will Reg Guide 1.59, Rev. 3, be issued? It's in the pipeline. It's in the process. You know, they work it through the process. We don't have any immediate need for it right now. So, I'm not --

CHAIR KIRCHNER: Right.

MEMBER HALNON: It doesn't raise to my level of concern that it's not going to get issued someday, as needed.

CHAIR KIRCHNER: So, then, you will make a modification to this?

MEMBER HALNON: Correct. I've got a bit

of rewrite to do. And the conclusion will be that there's no further recommendations on Chapter 2.

CHAIR KIRCHNER: So, any further input or questions on Chapter 2?

MEMBER MARCH-LEUBA: I was saying just with my head, because both approaches seem to take us to the same place.

MEMBER HALNON: Right.

MEMBER MARCH-LEUBA: So, let's take the easy one.

CHAIR KIRCHNER: Okay. Thank you for that one.

MR. BETANCOURT: Chair --

CHAIR KIRCHNER: Yes, go ahead.

MR. BETANCOURT: Chair Kirchner, the court reporter is still on right now and is recording this; for instance, just got what Mike read into the -- do you want the court reporter to continue to stay on during this portion?

CHAIR KIRCHNER: Yes.

MEMBER HALNON: For Chapter 13, we have a similar process.

CHAIR KIRCHNER: We have a similar process.

MR. BETANCOURT: Okay.

CHAIR KIRCHNER: Yes. No, I appreciate that.

MR. BETANCOURT: So, just let us know when we can cut the court reporter free.

CHAIR KIRCHNER: After Chapter 13 I would recommend.

MR. BETANCOURT: All right. Thank you.

CHAIR KIRCHNER: Yes.

So, with that, we'll turn, next, to Chapter 13, Greg.

MEMBER HALNON: Okay. Chapter 13 had to do with -- it's titled Conduct of Operations. The only one issue with that was -- and again, it was part of the delta review, if you will, between the COL, I mean, the design certification and, well, I guess it was a COL. And we have it for the SDAA.

In the COL, there was a specific statement relative to the plant, their technical guidelines. They're called different things, but, basically, it's the technical guidelines on how to -- whoa, the basis document for your EOPs and how you respond to accidents and casualties. They've removed that specific statement in the SDAA.

So, in thinking about that, it was important for me to see that, for an nth of a kind reactor, we were going to have a basis for responding to casualties, such that the very basis for part of the safety argument for these reactors is the human interface to the reaction to casualties, transients, and other things.

And, yes, we take much credit for this passive nature of these from a safety perspective, but there is still a last line of defense of the operators required to respond to accidents and other items. So, because we're taking credit for that portion of it, it was important that I see some consistency in how SDA Reactor No. 1 equals SDA Reactor No. 2, and put an nth onto that, such that all the reactors that we say are using this design certification are designed to that design. It is consistently applying that same level of response, so that we can say that the same level of safety is being applied.

I didn't say that real clearly, but I'm trying to say that, for an nth of a kind, at least in my opinion -- and this is the argument that we mentioned earlier today -- we're going to have in the future, what are the boundaries of an nth of a kind? And I can see that the operating procedures, normal operating procedures, don't need to be absolutely consistent, because you'll have different business issues and siting, and other things.

But the response to an accident from an EOP perspective, I can see that should be based on a consistent document with the plant technical guidelines.

Now, there's a lot of ways of doing that. The staff response to that question, it's long and it's got a lot of regulations and Reg Guides and other things, and NUREGs in it. But the bottom line is that I think we can still get there. This SDA, it may be early to be requesting that, and maybe that's part of the basis of why they took it out of the SDA SAR.

But we're going to have to be diligent in the COL process to ensure that we drive that consistency home, if we're crediting any kind of operator action to be part of the safety defense-in-depth aspect.

In the COL, Dennis wrote the letter, and he made that --

MR. SNODDERLY: The DCA.

MEMBER HALNON: I'm sorry -- the DCA. He wrote the same statement. And Mike may be aware of what Dennis wrote. He said the same thing: the COL, going to have to see the plant technical guidelines and how they're applied, and how it's consistently going to drive into the EOPs.

I think that's the same point we need to make here to be consistent with our previous position when we do get a Combined Operating License, because both of these are putting the onus on the owner of that COL to develop the procedure generation package, or the procedures. We're going to have to be diligent to make sure that that consistency is there.

So, I think, for ourselves, we should be consistent with our previous position, which I agree with. We don't need to see it here. We can see it down the road on the COL. But we need to drive that point home.

And again, my letter was written as if I need to have this open question answered, and they answered it.

Mike, I'm not sure if you want to read that whole thing in, but it's --

MR. SNODDERLY: Yeah, I'll paraphrase.

MEMBER HALNON: Okay.

MR. SNODDERLY: So, Member Halnon asked for a roadmap, and the staff responded with the roadmap. And the staff summarized the roadmap as containing five parts: regulatory requirements, Standard Review Plan guidance, expectations for SDAA and COLA submittals, the procedure generation package, and the construction and inspection program. And so, I'll just briefly summarize each of those pieces of the roadmap.

MEMBER HALNON: So, before you go, this is how we will see the consistency of the technical guidelines being driven into the EOPs and the operating procedures.

Go ahead, Mike.

MR. SNODDERLY: Thank you, Member Halnon.

The staff, first, referred to the regulatory requirements in 10 CFR 52.79, which requires managerial and administrative controls to be used to ensure safe operation, in accordance with Appendix B to 10 CFR Part 50 requirements.

Then, they also referenced 10 CFR 52.79(29), plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of SSCs, structures, systems, and components, and plans for coping with emergencies other than the plans required by 52.79(a)(21).

Then, there's also 10 CFR Part 50, Appendix B, Criteria 5 and 6, which establish criteria for the development, approval, and control of procedures for all activities affecting quality.

And then, they refer, also, to the requirements of 10 CFR 50.34(f)(2)(ii), contents of applications, technical information, and additional (audio interference) related requirements that establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts.

The second part of the roadmap was NRC staff review procedures, including the Standard Review Plan, Chapter 13, Conduct of Operations, and, more specifically, Section 13.521, Operating Emergency Operating Procedures, Revision 2, March 2007.

There's also the guidance in NUREG-0711, Human Factors Engineering Program Review, Section 9, Procedure Development.

Procedures are integral to an overall HFE program and should be developed and implemented using accepted HFE principles. The NRC reviews procedures to confirm that the applicant's procedure development program incorporates HFE principles and criteria.

The third tier of the roadmap would be

expectations for the SDAA and COLA submittals.

At the Standard Design Approval Application stage, the staff reviews COL action items for procedures. Plant procedures include administrative operating procedures, emergency operating procedures, as well as maintenance and other procedures for safety-related activities.

The COL applicant is responsible for these types of procedures. The staff's review is focused on the evaluation of COL action items pertaining to procedures. And the staff provides an example. COL item 13.5-5 addresses EOPs.

An applicant that references the NuScale power plant US460 standard design will provide a plan in the development, implementation, and control of emergency operating procedures, including preliminary schedules for preparation and target dates for completion. Then, additionally, the applicant will identify the group within the operating organization responsible for maintaining these procedures.

COL applicants or COL holders are required to develop procedures that are plant-specific.

And then, the fourth tier of the roadmap would be procedure generation package. Information about EOP development and implementation is supplemented for NRC staff review via the procedures generation package, or PGP. The PGP must be submitted for NRC review no later than three months before formal operator training on EOPs begins.

The procedure generation package contains the following, in accordance with SRP's Chapter 13.521:

Plant-specific technical guidelines. These may or may not reference the general technical guidelines.

Plant-specific writer's guide that details the methods to be used by the applicant in preparing the EOPs, based on the plant-specific technical guidelines.

A description of the verification and validation programs for EOPs and a description of the program for training operators on the EOPs.

And finally, the staff refers in their roadmap to the construction and inspection program. The NRC staff verifies the technical adequacy of the COL holder's operating procedures through the construction and inspection program.

Inspection procedures used by the staff include the following three inspection procedures:

42401 on plant procedures.

Inspection Procedure 42453 on operating procedures and inspections.

And finally, Inspection Procedure 42454 on emergency procedures.

And that concludes the staff's response to the Chapter 13 issue. And as I said, the entire response will be included as part of the transcript to this meeting.

And with that, I turn it back over to Lead Member Halnon for Chapter 15.

MEMBER HALNON: Thanks, Mike.

And I guess you can now see why I asked for a roadmap on how to get from here to there. And it doesn't ever really specifically require their technical guidelines.

However, the staff, through this, and through the fact that, at first, those regulatory requirements for procedures that the review guidance has to look at procedures, there's expectations for those procedures to be part of the applications. The procedure generation package is developed and reviewed and looked at by the staff, and then, it's inspected, once it's in place.

All those are the right elements. The only problem is now the onus to make sure that they're consistent and will develop and fit a definition of nth of the kind, if that's the argument we're going to have, is on the staff, because they're all plant-specific. So, you can't ask Plant No. 2 to go check with Plant No. 35, or vice versa, to see if their procedures are consistent.

Maybe they don't have to be. Maybe they can have some certain level of deviation. However, I think that's kind of one of those future things we're going to be discussing as a Committee relative to this.

So, this will get you there. It does allow it to be, and there's enablers in there to be, consistent. And I think it's adequate for the stage that we're at right now.

This is the first one we're doing. So, we're going to learn some more from it and we will see where we go.

But the bottom line is that the requirements are all there and the roadmap is, in my mind, adequate to ensure that there's at least a level of thought being put into it.

So, I asked for the roadmap, or verbally in the letter I wrote that recommended that the COL item that is referenced in the SDA be identical to the

DCA, the same as the DCA, where Dennis made that statement. I've got to revise that to be consistent with what we did before and not make a recommendation that the SDA needs to be changed, but to focus more on the roadmap and making sure that we get a chance to look at it in its COL stage, so that we can ensure that there's a consistent approach to that level of safety to the plant.

So again, homework.

CHAIR KIRCHNER: Okay. Thank you.

All right. And with that, now we'll regress a little bit, so to speak, at least in terms of numerical order, and go to Chapter 10 and Matt Sunseri.

Matt?

MEMBER BIER: Well, now do you want to release the court reporter?

CHAIR KIRCHNER: Yes, I think we could. Yes.

At this point, we can release the court reporter. But thank you for your service today.

(Whereupon, the above-entitled matter went off the record at 4:05 p.m.)

Status Update on Computer Code and Model Development for non-LWRs

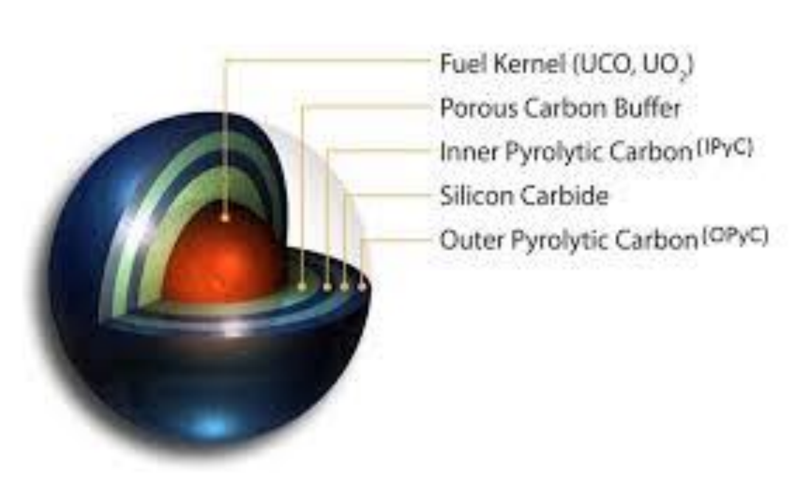
Kimberly A. Webber, Ph.D.

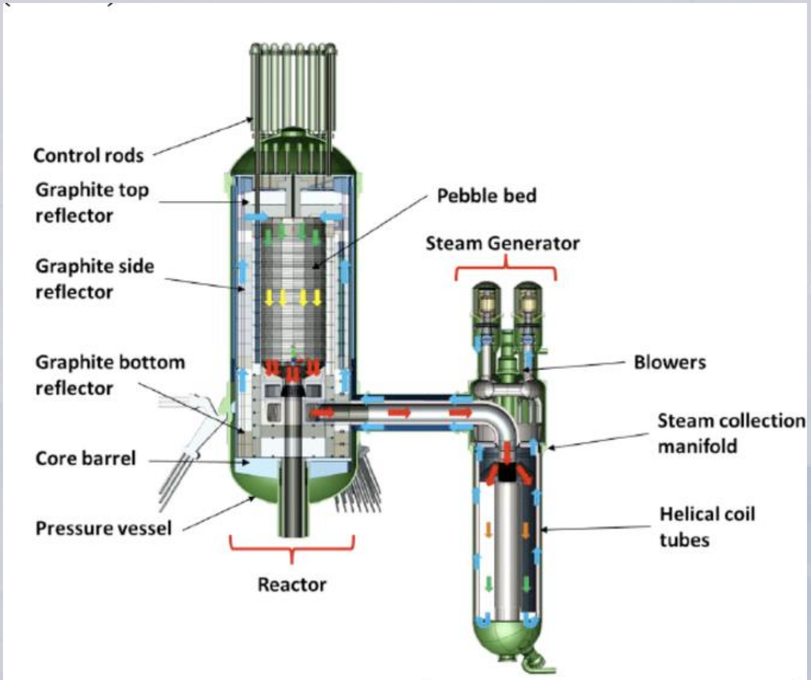
Director, Division of Systems Analysis

Office of Nuclear Regulatory Research

Agenda

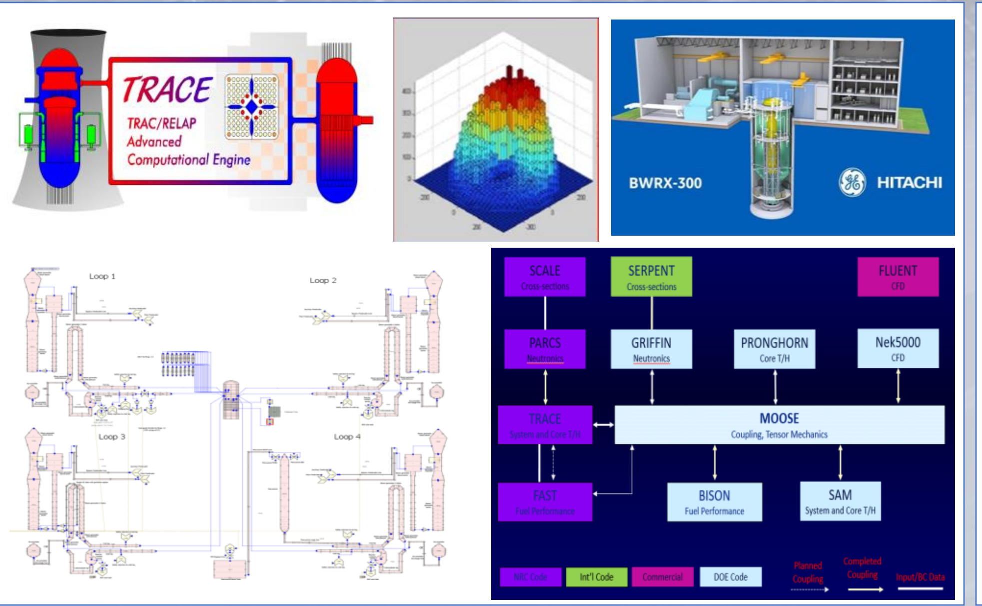
1. Overview
2. Plant Systems Analysis
3. Fuel Performance Analysis
4. Severe Accident Progression
5. Consequence Analysis
6. Licensing and Siting Dose Assessment
7. Nuclear Fuel Cycle Analysis
8. Conclusion

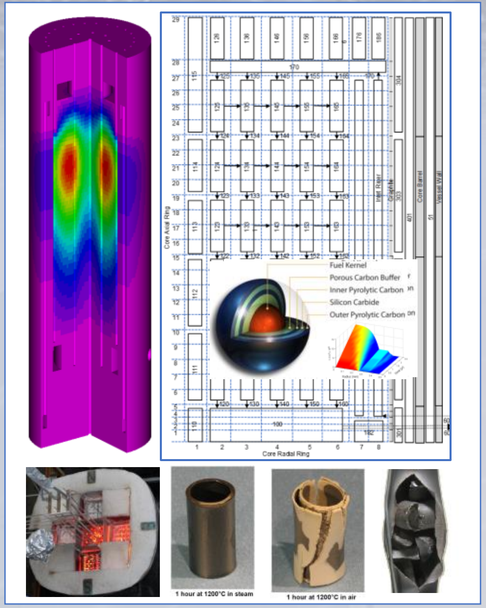


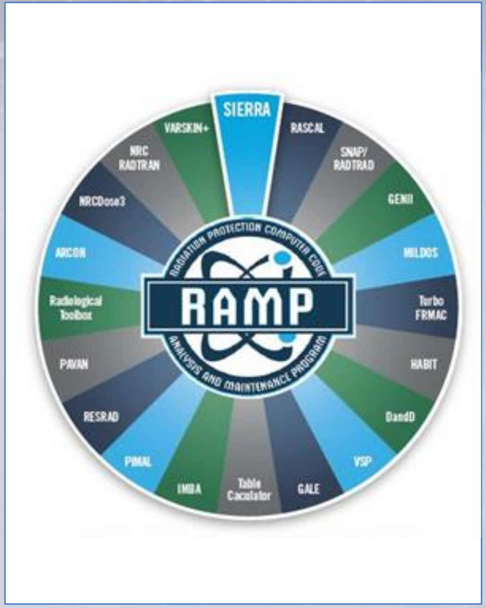


Xe-100 Reactor System Configuration (Courtesy of X-energy, with permission)

Division of Systems Analysis (DSA) Branches

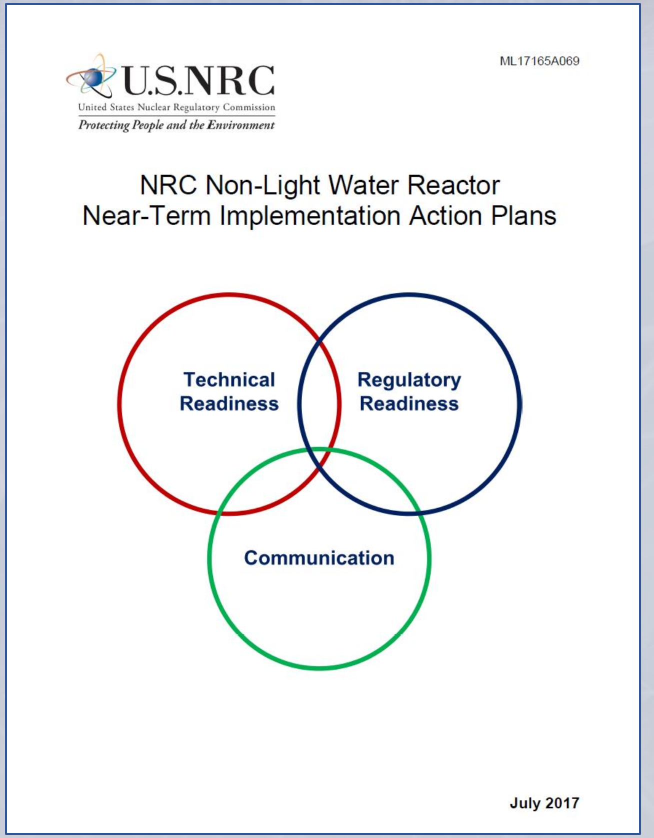






Code and Reactor Analysis Branch – I (CRAB-I)	Code and Reactor Analysis Branch – II (CRAB-II)	Fuel & Source Term Code Development Branch (FSCB)	Accident Analysis Branch (AAB)	Radiation Protection Branch (RPB)
Chris Hoxie	Kenneth Armstrong	Hossein Esmaili	Luis Betancourt	John Tomon
IAP Volume 1 (Incl. Steve Bajorek)		IAP Volumes 2, 3 & 5	IAP Volume 3	IAP Volume 4

Integrated Action Plan (IAP) for non-Light Water Reactors


ML17165A069

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NRC Non-Light Water Reactor
Near-Term Implementation Action Plans

July 2017

[ML17165A069](ML17165A069)

Near-Term Implementation Action Plan

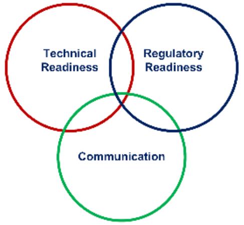
- Strategy 1: Knowledge, Skills, and Capacity
- Strategy 2: Analytical Tools
- Strategy 3: Flexible Review Process
- Strategy 4: Industry Codes and Standards
- Strategy 5: Technology Inclusive Issues
- Strategy 6: Communication

NRC Code Development Reports

RES

Revision 1
January 31, 2020

Approach for Code Development in Support of NRC's Regulatory Oversight of Non-Light Water Reactors

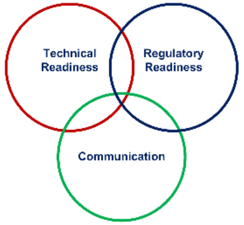


Introduction

[ML20030A174](ML20030A174)

Revision 1
January 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 1 – *Computer Code Suite for Non-LWR Plant Systems Analysis*

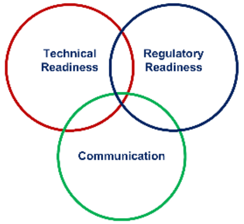


Volume 1

ML20030A176

Revision 1
January 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 2 – *Fuel Performance Analysis for Non-LWRs*

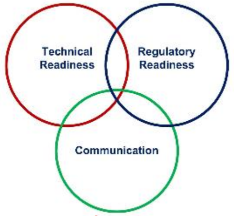


Volume 2

ML20030A177

Revision 1
January 31, 2020

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*

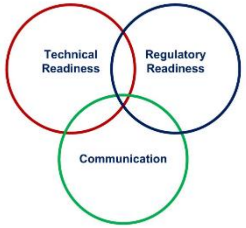


Volume 3

ML20030A178

Revision 1
March 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 4 – *Licensing and Siting Dose Assessment Codes*

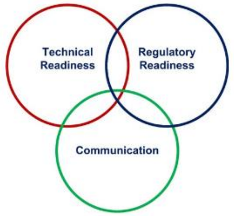


Volume 4

ML21085A484

Revision 1
March 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*



Volume 5

ML21088A047

U.S.NRC
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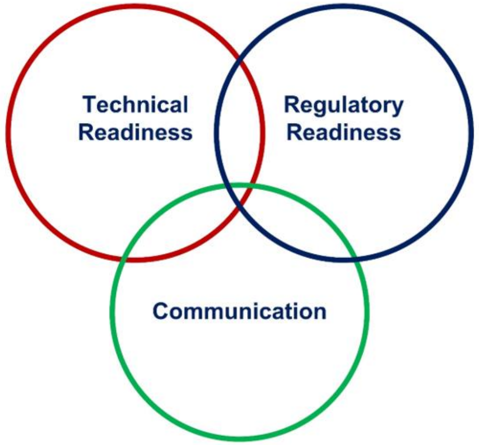
Protecting People and the Environment

Status Update on Computer Code and Model Development for Non-Light-Water Reactors



Office of Nuclear Regulatory Research
Division of Systems Analysis

March 2024

ML24069A003

Recent History of Interactions with ACRS (1/2)

- Many meetings held between DOE, NRC and ACRS over the past several years to discuss codes development efforts to support industry and NRC licensing of non-LWRs
- ACRS Conclusions
  - ✓ Significant effort by the staff to develop non-LWR code analysis capability substantially increases the readiness of the staff, promoting expeditious reviews
  - ✓ Importance of independent capability for confirmatory analyses
  - ✓ Reference plant model approach useful to assess adequacy of codes and assess data gaps
  - ✓ Consolidating radiation protection codes is comprehensive and workable
  - ✓ Flexible and workable strategy to address fuel cycle code development needs
  - ✓ Importance of code validation
  - ✓ Importance of developing staff expertise

Recent History of Interactions with ACRS (2/2)

RES

- ACRS Recommendations
  - ✓ Seek simplified solutions when adequate for the problem
  - ✓ Perform pilot studies to illustrate analysis capability
  - Scale down the level of effort of licensing review proportionately as the hazard decreases

Conclusions

RES

Completed

- Non-LWR Code Development Reports
- Reference Plant Models
- SCALE/MELCOR Demonstration Public Workshops
- MACCS assessments and updates
- Code Assessment Reports for Metallic and TRISO Fuels
- Training on BlueCRAB Codes

Next steps

- New and Updates to Existing Reference Plant Models
- Verification and Validation (V&V) Report for Systems Analysis
- Assessment of MACCS capabilities to model physiochemical transformations during atmospheric dispersion
- Development/consolidation of Radiation Protection Codes for non-LWR analysis
- Fuel Cycle Demonstration Project Public Workshop for Molten Salt Reactor

Historical Content on Non-LWRs and Code Development Activities

RES

Title	Date	Material	ML
Briefing to ACRS Thermal-Hydraulic Subcommittee by DOE	Aug. 21, 2018	Transcript	18254A164
Briefing to ACRS Thermal-Hydraulic Phenomena Subcommittee by DOE	Nov. 16, 2018	Transcript	18340A016
Briefing to ACRS Future Plant Designs Subcommittee by NRC Staff	May 1, 2019	Transcript	19143A120
Subject: Review of Advanced Reactor Computer Code Evaluation	Nov. 4, 2019	ACRS Letter	19302F015
Subject: RES Response to ACRS Letter Dated Nov. 4, 2019	Jan. 31, 2020	RES Response	20030A172
Subject: Biennial Review and Evaluation of NRC Safety Research Program	Apr. 13, 2020	ACRS Letter	20100F066
Briefing to ACRS Future Plant Designs Subcommittee by NRC/RES Staff Subject: Non-LWR Code Development, Volume 4, "Licensing and Siting Dose Assessment Codes"	Sep. 22, 2020	Agenda Transcript	20255A222 20307A524
Briefing to ACRS Future Plant Designs Subcommittee by NRC/RES Staff Subject: Non-LWR Code Development, Volume 5, "Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle"	Dec. 1, 2020	Agenda Transcript	20328A290 21036A180
682nd meeting of ACRS Subject: Review of Two Volumes of Evaluations of Computer Codes to be Used for Analyses of Advanced Non-LWR Reactors	Feb. 3-5, 2021	Agenda Transcript ACRS Letter Staff Response	20351A370 21055A742 21053A024 21088A409
Briefing to ACRS Future Plant Designs Subcommittee by NRC Staff Subject: Integration of Source Term Activities in Support of Advance Reactor Initiatives	Feb. 17, 2022	Agenda Transcript ACRS Letter	22026A359 22060A171 22069A083

Readiness for Advanced Reactor System Analysis

“Volume 1 and BlueCRAB”

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Senior Level Advisor for Thermal Hydraulics

Division of Systems Analysis

Office of Nuclear Regulatory Research

Introduction / Agenda

- Background information on Volume 1 Approach
  - Intended applications
  - Why “BlueCRAB” ?
- Verification & Validation (V&V) Report
  - Content
  - Validation Status
- Reference Plant Development
  - General approach & status
  - Sample results
- Summary and Next Steps

Acknowledgements

Development of BlueCRAB, Reference Plant Models, and the V&V Report is result of a coordinated effort between NRC, INL, and ANL.

Contributors include:

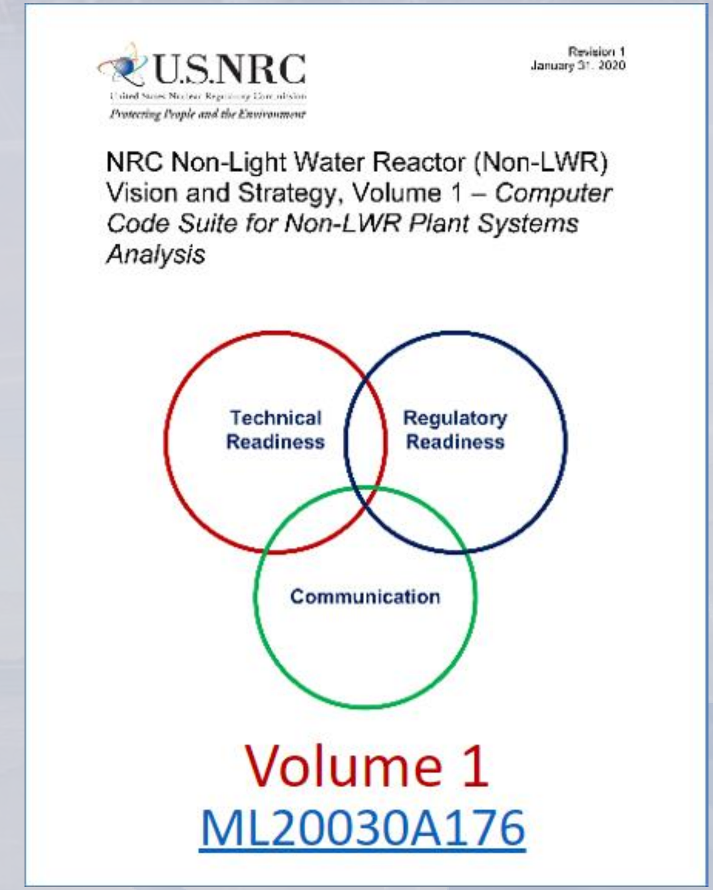
NRC: *Kenneth Armstrong, Stephen M. Bajorek, Matthew Bernard, Andrew Bielen, Joseph Kelly, Cole Takasugi, Nazila Tehrani, Jason Thompson, Tarek Zaki*

ANL: *Rui Hu, Travis Mui, Zhiee Jhia Ooi, Emily Shemon, Gang Yang, Ling Zou*

INL: *Namjae Choi, Joshua Hanophy, Logan Harbour, Jackson Harter, Joshua Hensel, Mustafa Jaradat, Javi Ortensi, Cody Permann, Stefano Terlizzi*

Volume 1 Intended Applications

- Volume 1 of the “Implementation Action Plan”:
  - Define codes for system analysis for all non-LWR technologies.
  - Reviewed PIRTs to identify important phenomena, scenarios and potential knowledge gaps
- Intended Applications & Uses:
  - Steady-state conditions with power, temperature and velocity distributions.
  - Accident analysis for scenarios not resulting in core disruption including loss-of-flow, loss-of heat sink, LOCA, reactivity insertions, heat pipe failure, etc.
  - Staff education: “How should the machine perform ?”


U.S.NRC
Protecting People and the Environment

Revision 1
January 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 1 – *Computer Code Suite for Non-LWR Plant Systems Analysis*

Technical Readiness
Regulatory Readiness
Communication

Volume 1
ML20030A176

Advanced Reactor Landscape

Small Modular Light Water Reactors

Pressurized Water Reactors
- NuScale US600
- NuScale US460
- Holtec SMR-300 *
- Westinghouse AP300*

Boiling Water Reactor
- GEH BWRX-300*

Liquid Metal Cooled Fast Reactors (LMFR)

Sodium-Cooled
- TerraPower/GEH (Natrium)*
- GEH PRISM (VTR)
- ARC Clean Technology *

Lead-Cooled
- Westinghouse
- Columbia Basin
- Hydromine

High-Temperature Gas-Cooled Reactors (HTGR)

TRISO Fuel
- X-energy (Xe-100)*
- Framatome
- StarCore
- MIT

Other:
- General Atomics (EM2) *
- General Atomics*

Molten Salt Reactors (MSR)

Liquid Salt Cooled
- Kairos *
- Kairos (Hermes | RTR)

Liquid Salt Fueled
- Terrestrial *
- TerraPower (MCFR)
- Southern (TP MCRE) | RTR
- ACU | RTR
- Elysium
- ThorCon
- Muons
- Flibe
- Alpha Tech

Micro Reactors

Transportable
- Westinghouse (eVinci) *
- BWX Technologies
- X-energy
- Radiant *
- Ultra Safe Nuclear
- Alpha Tech
- Nano Nuclear Energy

Stationary
- UIUC / Ultra Safe | RTR*
- Oklo *

LEGEND

ARDP Awardees
- Demo Reactors
- Risk Reduction
- ARC-20

(thick black border) In Licensing Review

* Preapplication

RTR Research/Test Reactor

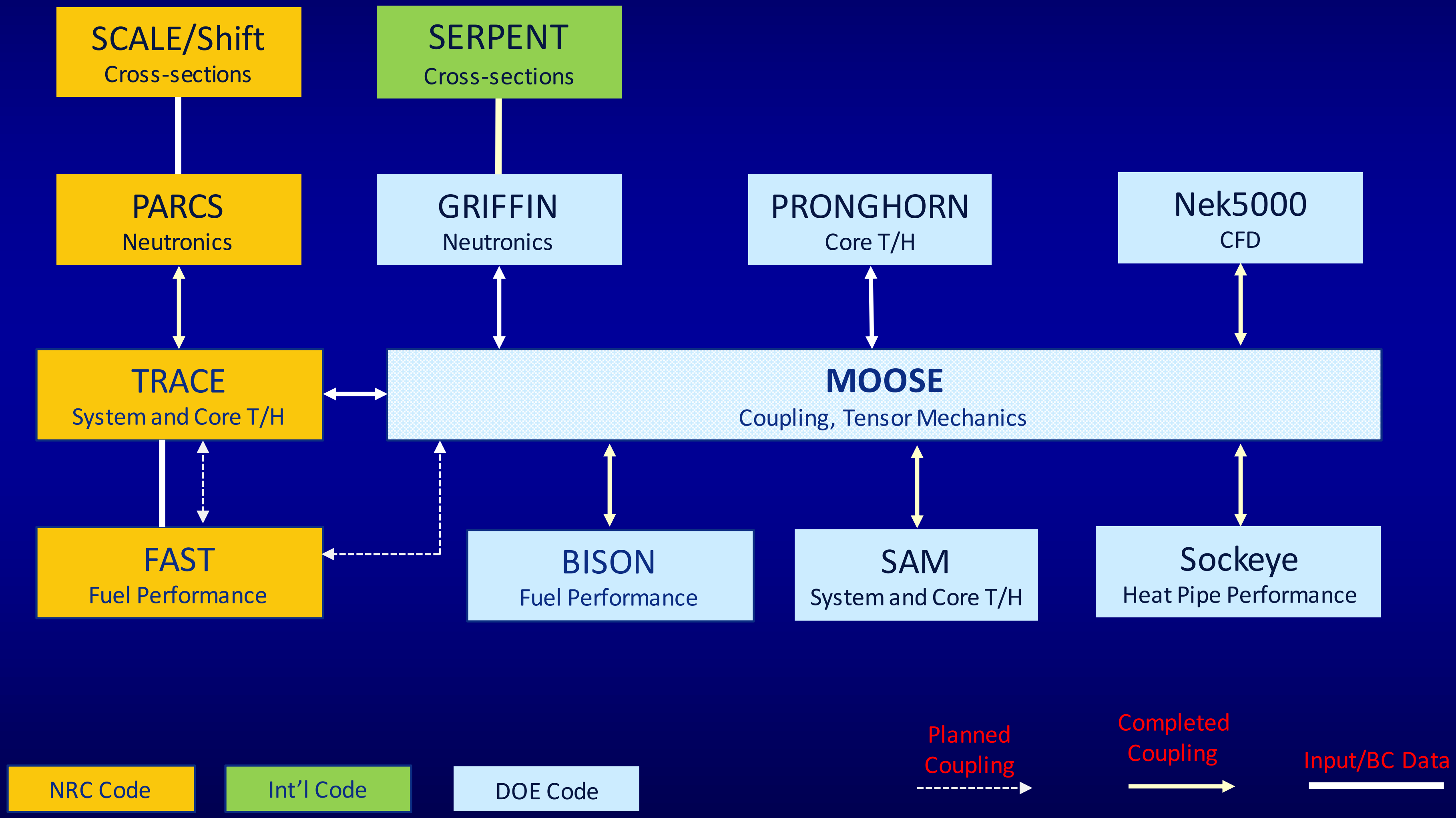
"Modeling Gaps" Identified by PIRTs

- 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)

↓

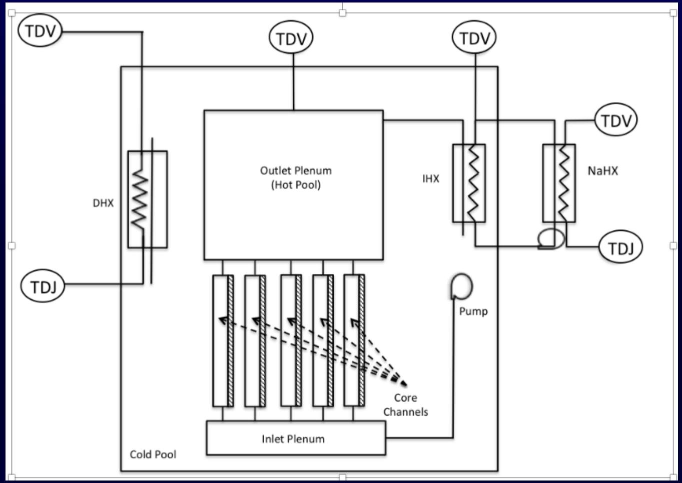
"Modeling Gaps in NRC Codes"

Comprehensive Reactor Analysis Bundle
BlueCRAB

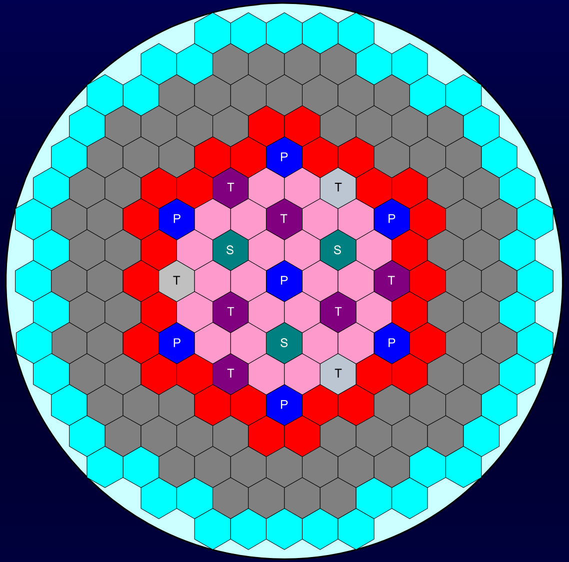


Multiphysics Coupling

SAM: System Level Thermo-Fluids



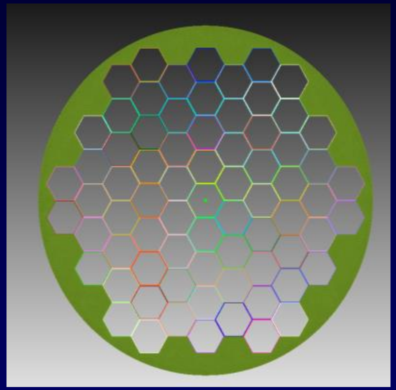
Griffin: Reactor Dynamics



Temperatures & Densities

Power

Temperatures



Displacements

Tensor Mechanics Module

Verification & Validation

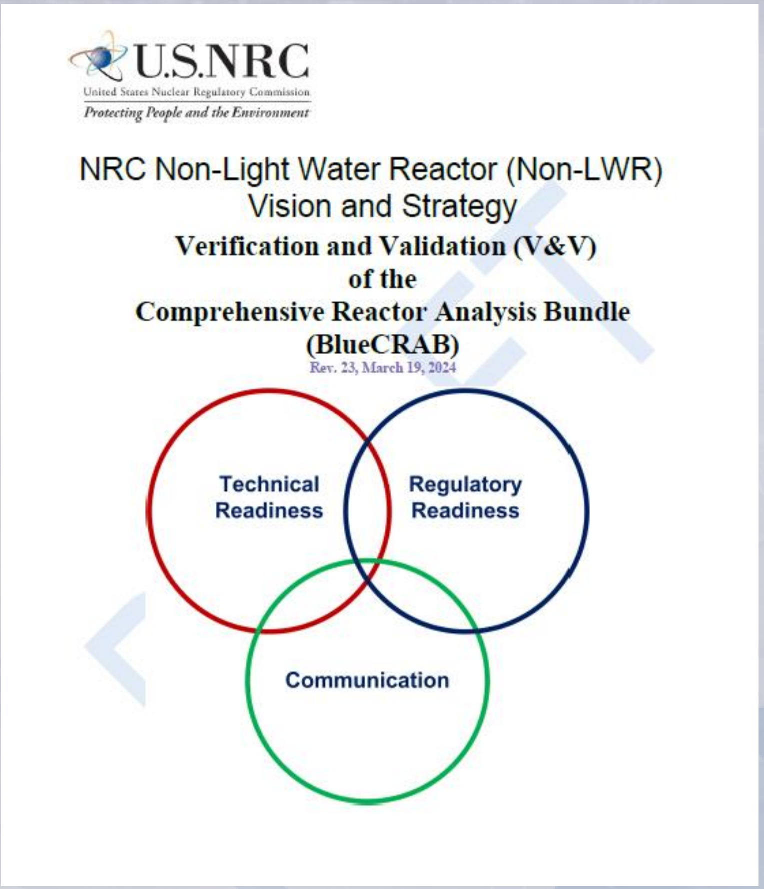
RES

- Verification & Validation (V&V) are vital components of the "Evaluation Model Development and Assessment Process" (EMDAP) as summarized in RG 1.203.

- Draft V&V Report for Volume 1 developed to
  - Document available PIRTs for each technology
  - Identify verification standards for each code
  - Cite the applicable validation for BlueCRAB codes by major technology
  - Help identify assessment and database gaps
  - Provide a quick reference on test facilities and benchmarks

BlueCRAB V&V Report Contents

- Brief description of BlueCRAB codes
- PIRTs & Scenarios
- Verification, including code coupling
- Validation by technology
  - Gas-cooled
  - Liquid metal
  - Molten salt
  - Microreactors
  - Neutronics
  - Components (heat pipes, local phenomena)
- Test and Benchmark Description & References



BlueCRAB V&V Report Contents

– Keyword is DRAFT. Some work is on-going. New data from university programs such as NEUP to be added.

Table 4 Validation for Gas-Cooled Reactors

Test	T	F	K	M	Code(s) Involved	Type	Design Type	Status	Validation Reference
HTTR	X		X		Griffin, BISON, Pronghorn	IET	HTGR	DOE-O	[5-1], [5-2] [5-37], [5-38], [5-39], [5-40]
HTTF	X				SAM	IET	HTGR	DOE-C	[5-3], [5-4], [5-25],[5-26] [5-28]
HTTF	X				Pronghorn, Nek	IET	HTGR	DOE-O	[5-34], [5-55]
HTR-10			X		Shift, Griffin	IET	PBMR	DOE-C	[5-5], [5-32]
HTR-10	X				SAM	IET	PBMR	DOE-P	[5-63]
HTR-PM	X		X		SAM, Pronghorn, BISON, Griffin	IET	PBMR	DOE-O	[5-6], [5-41] [5-42]
THTR-300	X		X		SAM, Pronghorn, BISON, Griffin	IET	PBMR	DOE-P	[5-7]
TAMU ΔP	X				SAM, Pronghorn	SET	PBMR MSPB	DOE-C	[5-8], [5-33]

Denotes model in the Virtual Test Bed:

[5-40] VTB, High Temperature Engineering Test Reactor (HTTR) Multiphysics Model, https://mooseframework.inl.gov/virtual_test_bed/htgr/httr/index.html

Placeholder for future reference

[5-7] TBD, Planned. (*Simulation depends on recovering the data and consideration of a thorium fuel cycle.*)

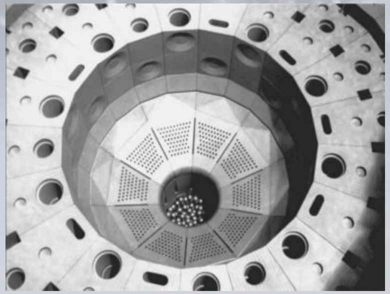
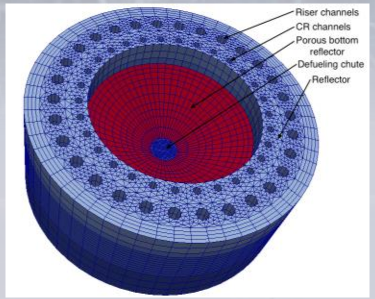
– Tables and highlights are intended to quickly show what assessment is complete, and what could be done.

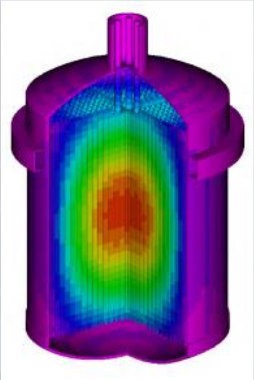
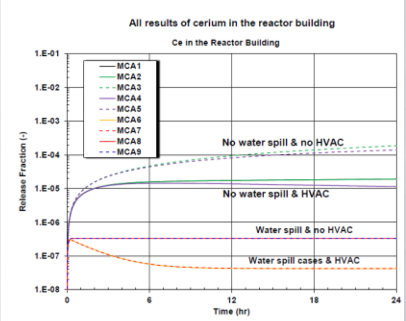
BlueCRAB V&V Report Comments

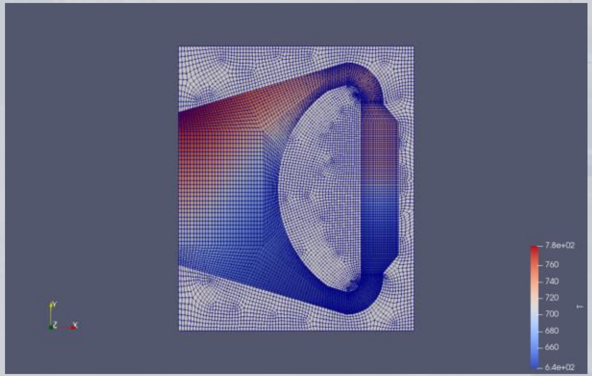
- At least some assessment has been completed for all technologies.
- Gas-cooled systems and Sodium liquid metal systems have received the most attention and have the most assessment.
- Molten fuel salt assessment is highly dependent on the MSRE (10 MWt). Scaling these data to other designs may be an issue.
- Heat pipe experimental data is available, but assessment is lagging. More work is necessary.
- Microreactor assessment will depend on prototypes currently planned or under construction.

Reference Model Development

- Reference Models - Generic representation of a design type, based on publicly available information.
- Scenarios "of interest" are selected (loss-of-flow, loss-of-heat sink, rapid reactivity insertion).
- Simulations performed to demonstrate code capabilities and *identify deficiencies before licensing reviews begin*.


HTR-10



MSRE



MCFR

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Reference Plant Status

Type	Reference Design	Accomplishments
GCR	HTR-PM (250 MWt)	2D Porous Media, Pebble Tracking & Equilibrium Core, RCCS
SFR	ABTR (250 MWt)	61 Chan Model with DRACS, Thermal Expansion, Doppler
MSR (cooled)	PB-FHR (320 MWt)	2D Porous Media, Pebble Tracking & Equilibrium Core, RCCS
MSR (fueled)	MSRE (10 MWt)	Neutron Precursor Tracking, Neutron Diffusion
MicroRx	~ SPR A	Heat Pipe Modeling, 3D Heat Conduction, Neutron Diffusion, Thermal Expansion
MicroRx	~ eVinci	Heat Pipe Modeling, 3D Heat Conduction, Neutron Transport

Other Available Reference Plants

Type	Reference Design	Comments
GCR	HTTF, PBMR-400	Benchmark Participation, Validation
SFR	FFTF	Benchmark Participation, Validation
MSR (fueled)	EVOL	Internal Circulation (i.e., requires ~ CFD)

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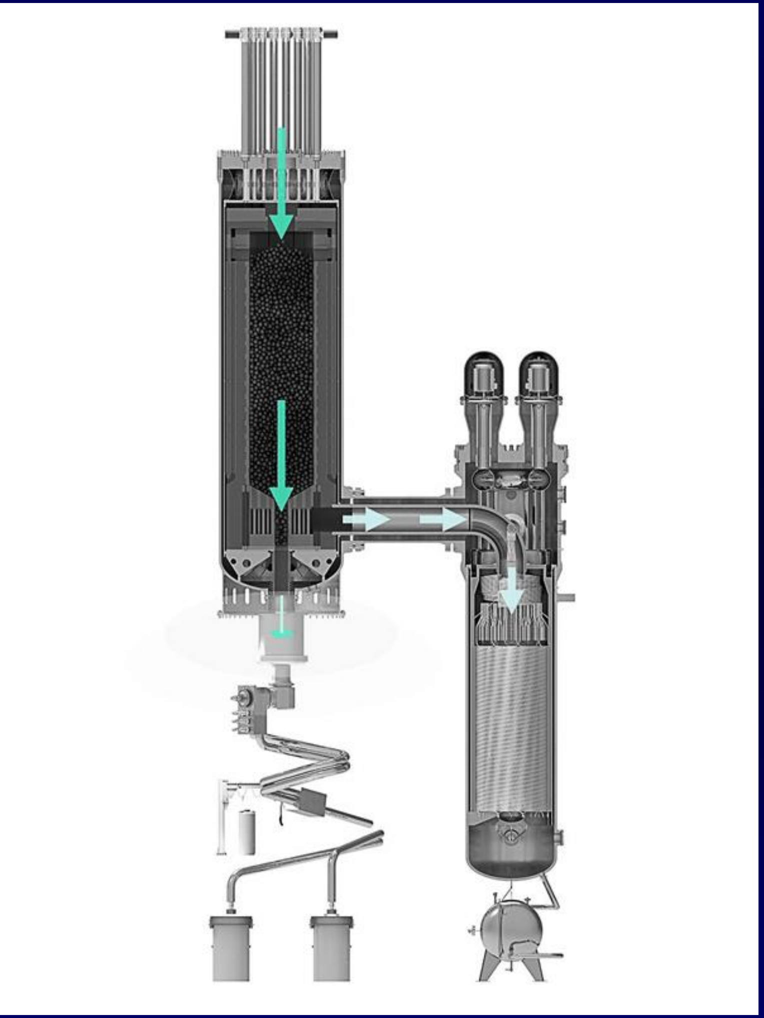
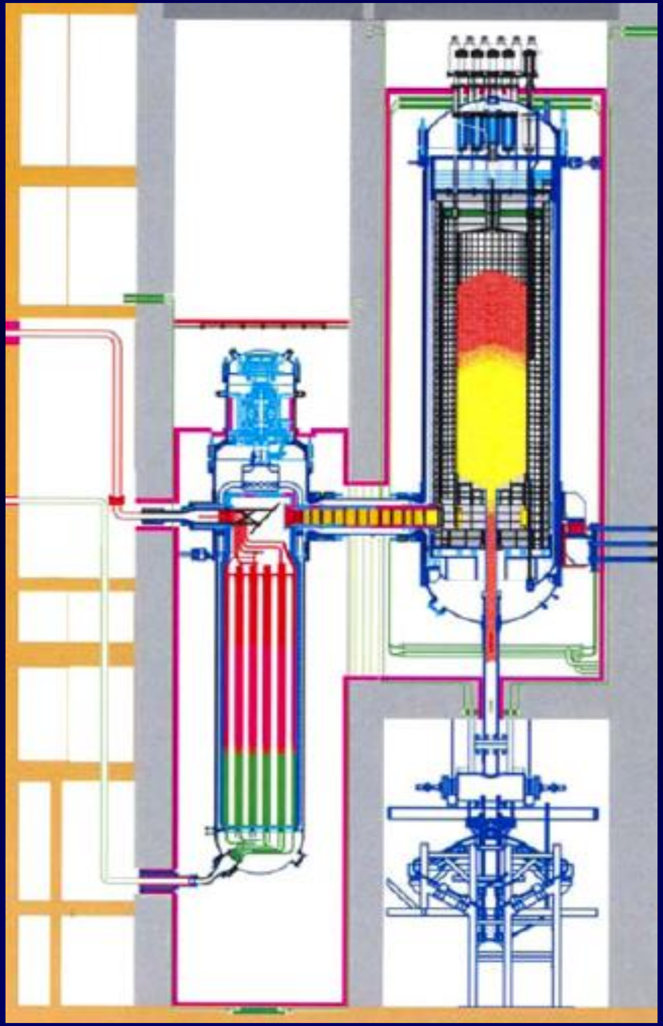
Gas-Cooled Pebble Bed Reference Plant: HTR-PM



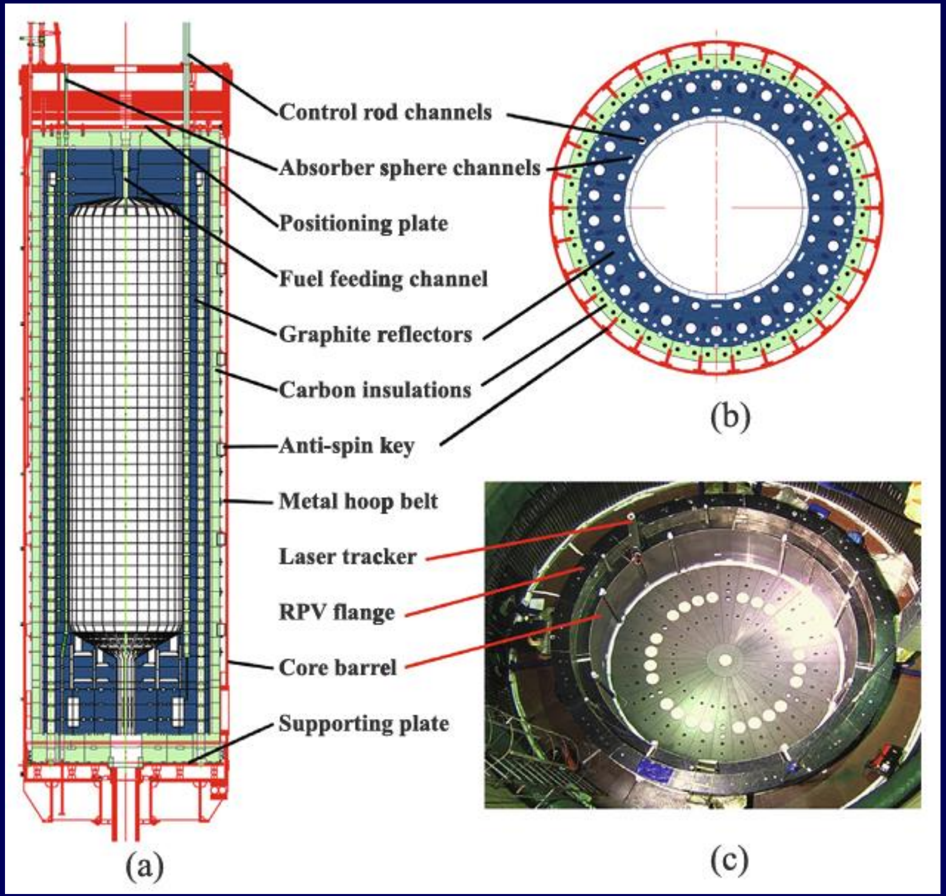
Xe-100 (200 MWt)



HTR-PM (250 MWt)

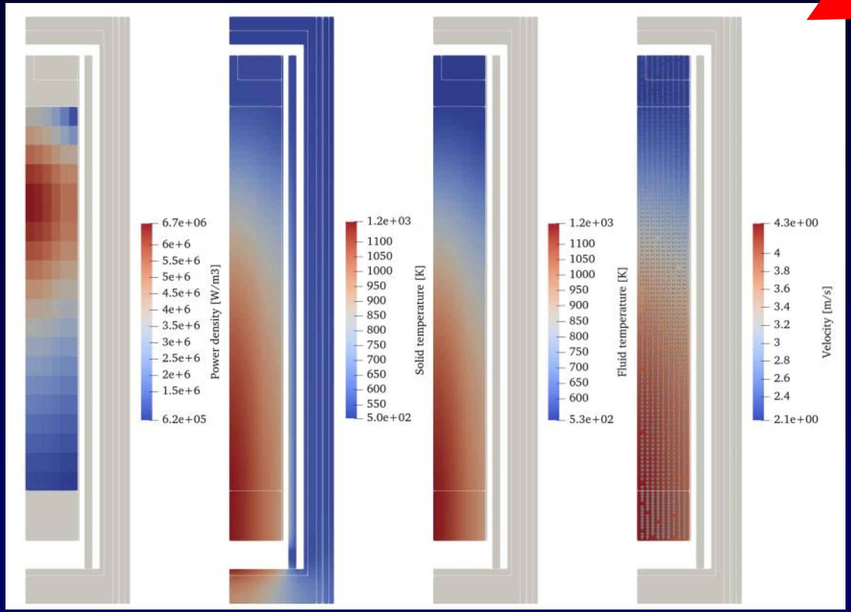
HTR-PM General Design

Parameter	Value
Core power [MWth]	250.00
Core inlet temperature [K]	523.15
Core outlet temperature [K]	1023.15
Core outlet pressure [MPa]	7.0
Pebble-bed radius [m]	1.50
Pebble-bed height [m]	11.00
Reflector outer radius [m]	2.50
Control rods channels	24
Reactivity Shutdown Channels	4
Barrel outer radius [m]	2.69
Bypass outer radius [m]	1.69
Vessel outer radius [m]	3.00
Number of pebbles	419,384
Pebble types	1 type
Avg. pebble packing fraction	0.61
Avg. number of passes	15
Avg. pebble residence time [days/pass]	70.5



HTR-PM Applications

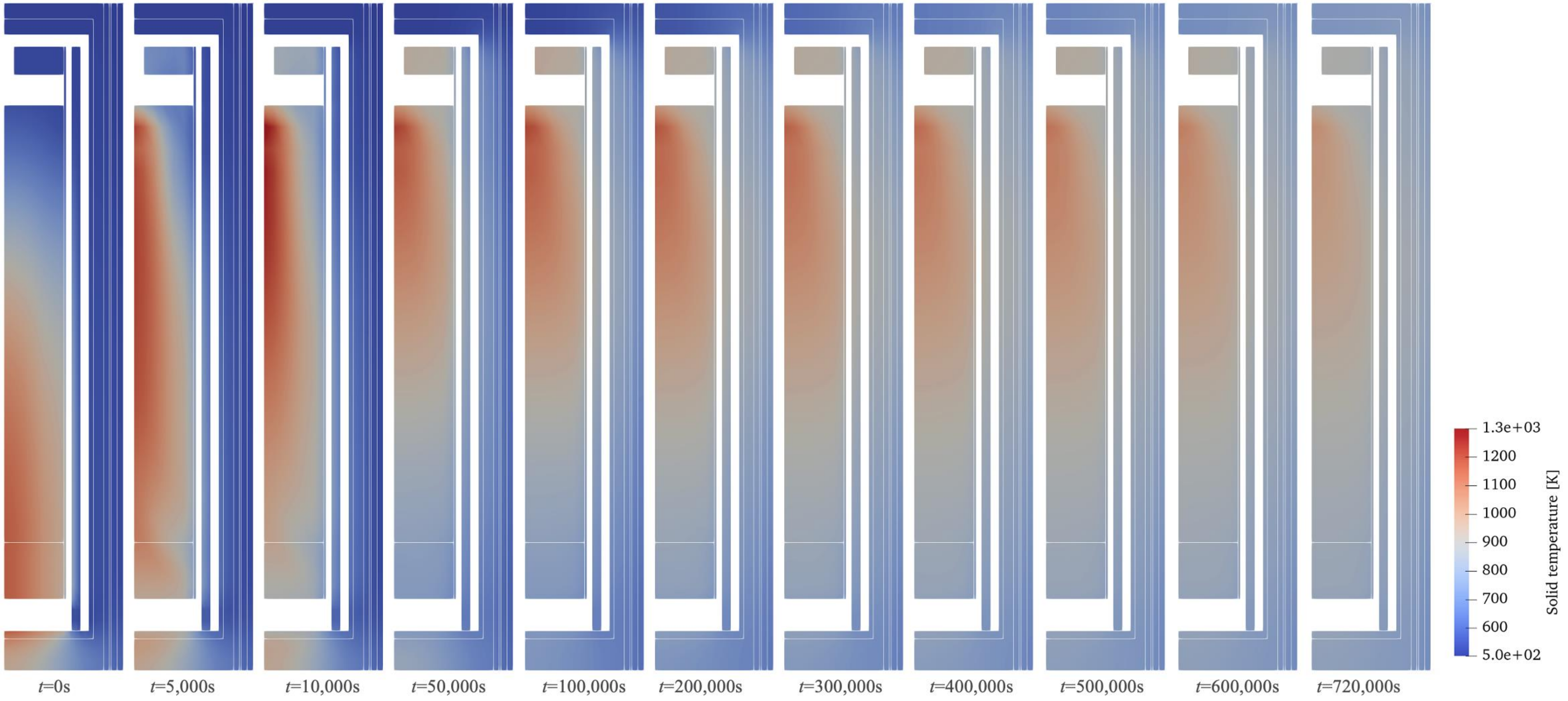
- Coupled SAM/Griffin multiapp, with 2D (r,z) porous media core & vessel with 1D loops. Includes air-cooled RCCS.
- Griffin used for pebble tracking, depletion, equilibrium core, and provides core axial & radial power, isotope distribution.
- Coupled model used for:
  - Steady-state temperatures & flow
  - Overcooling transient (reactivity insertion)
  - P-LOFC
  - D-LOFC
  - Small leak / LOCA (planned)



Power T_{Solid} T_{Fluid} Velocity

Results: Transient Pressurized Loss of Forced Cooling (PLOFC)

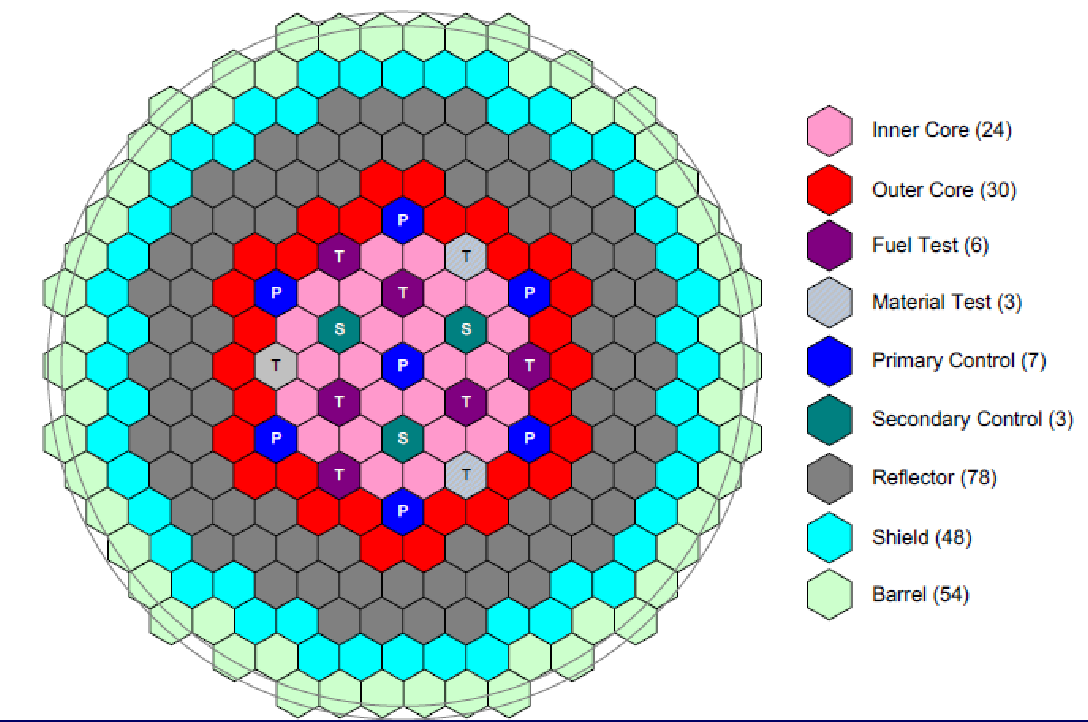
Solid temperature distributions

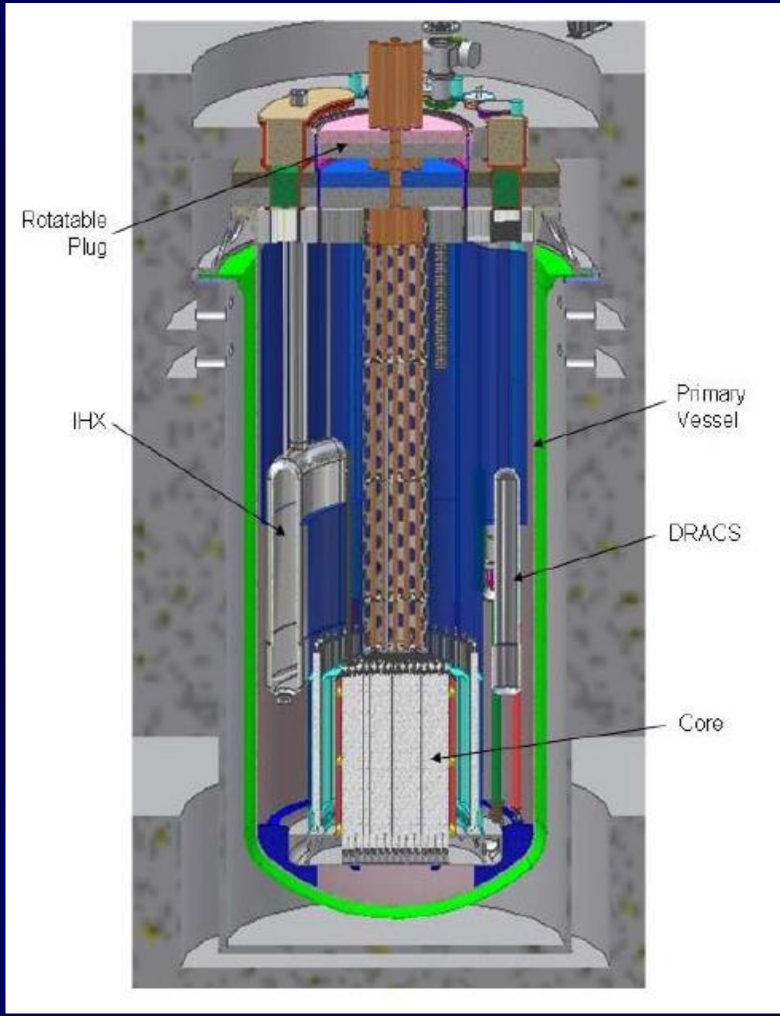


Credit: Rui Hu, et al. (ANL)

Sodium Fast Reactor Reference Plant (ABTR)

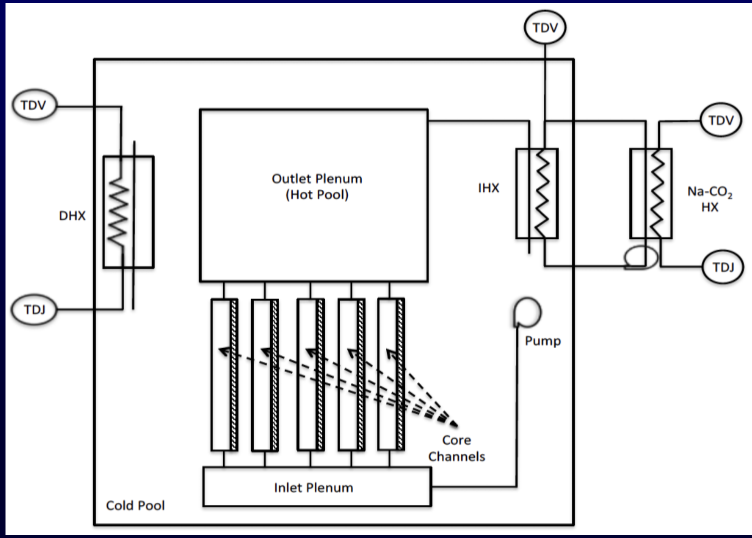
- Power: 250MWt, 95 Mwe
- Coolant: Sodium
- Temperatures: 355 °C/510 °C
- Reactor Vessel: 5.8 m diameter, 16 m height

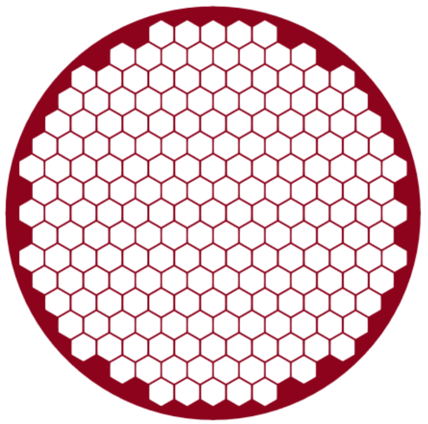


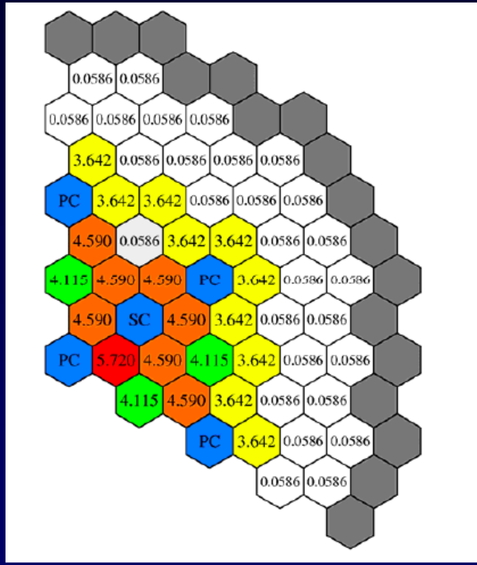


ABTR Model

- Reactor core: 61 channel representation
- Simplified intermediate loop, with two heat exchangers
- DRACS (DHX) is modeled
- Inlet plenum (cold pool), outlet plenum (hot pool), modeled with 0-D volumes
- A cover gas on top of the hot pool
- Reactivity components: Doppler, Axial fuel expansion, Sodium temperature and density, Radial support plate expansion.
- Thermomechanical model of support plate using BISON.
- Unprotected loss of flow (ULOF) scenario simulated.

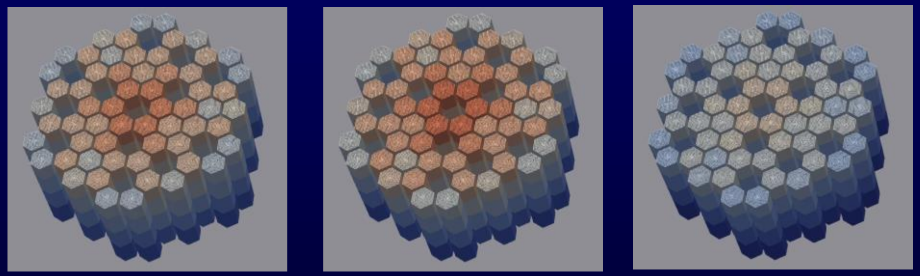


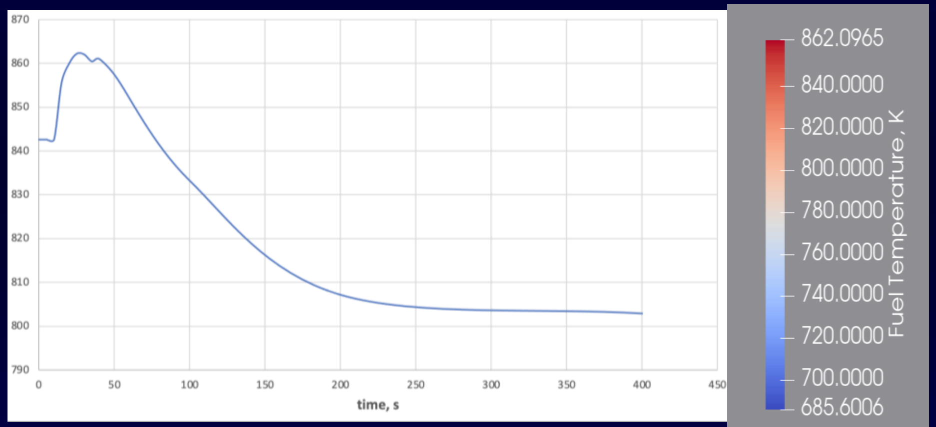




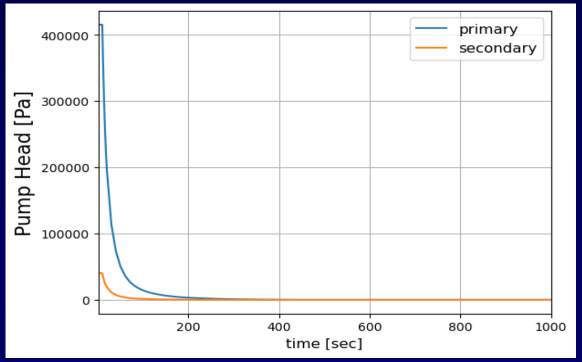
ULOF in ABTR

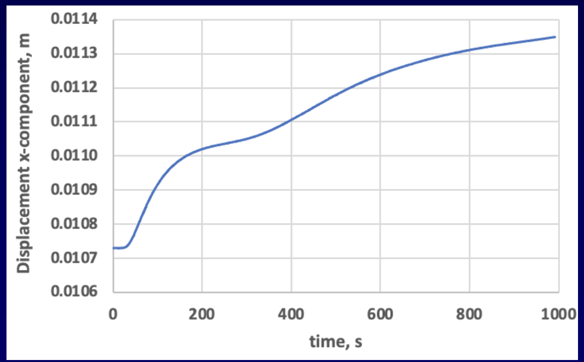
- ULOF is considered a beyond-design-basis accident . . .
- Power to both the primary and secondary coolant pumps is lost, and reactivity scram mechanisms assumed to fail.
- Mass flow rate decreases to zero due to pump head decreasing.
- Support plate displacement follows the inlet sodium temperature trend.
- Power decreases due to support plate thermal expansion and Doppler feedback

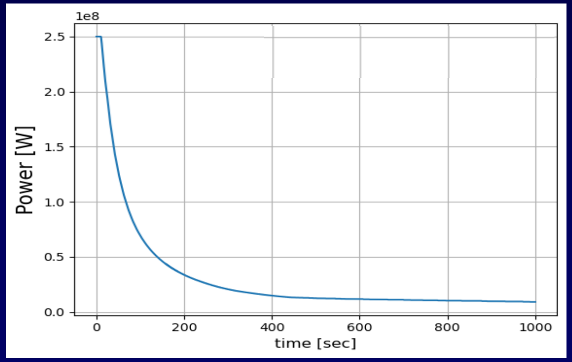




Credits: J. Ortensi et al (INL), R. Hu, et al. (ANL)

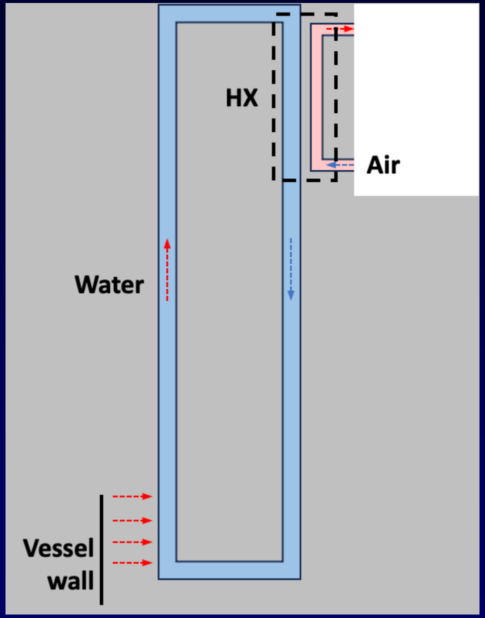


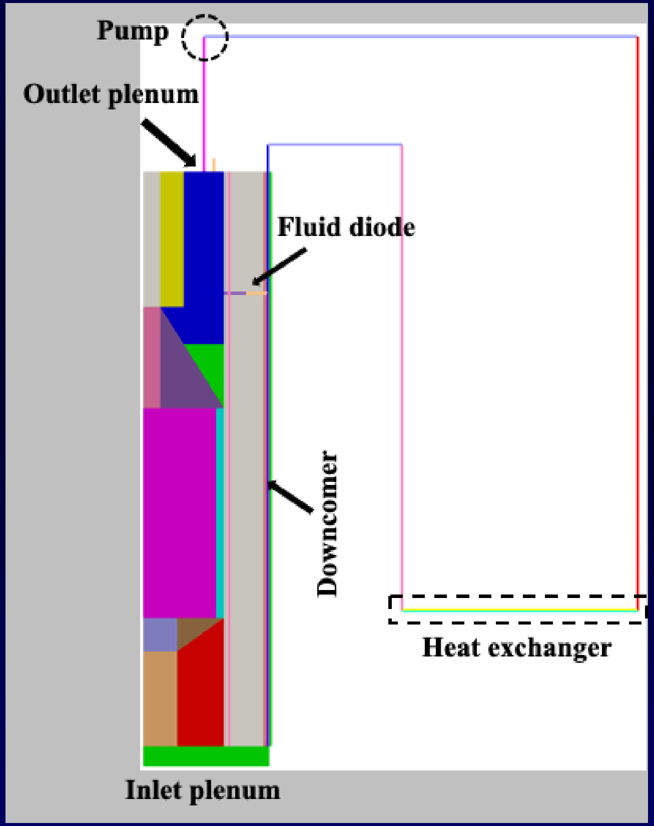




Molten Coolant Salt Reference Plant (PB-FHR)

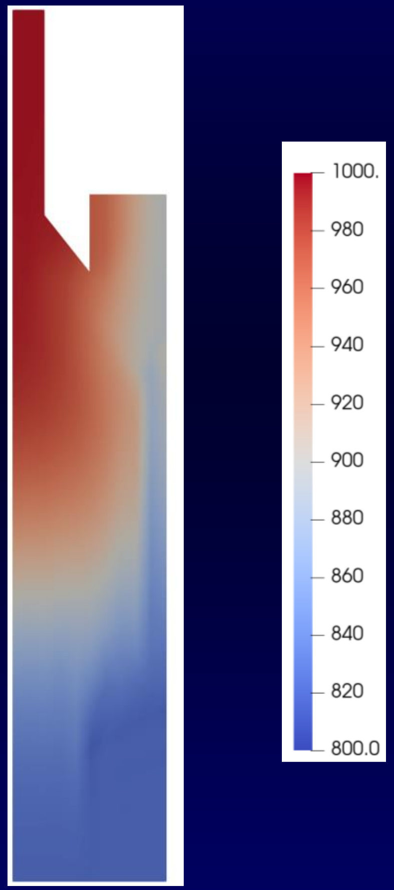
- Pebble Bed Fluoride High Temperature Reactor (PB-FHR)
  - 320 MWt
  - Coolant salt = FLiBe
  - 4 cm diameter (buoyant) pebbles
  - TRISO with 19.55% enrichment
  - Water cooled RCCS
  - UCO
  - 62.25 day transit
  - Tin = 550 C
  - Tout = 650 C



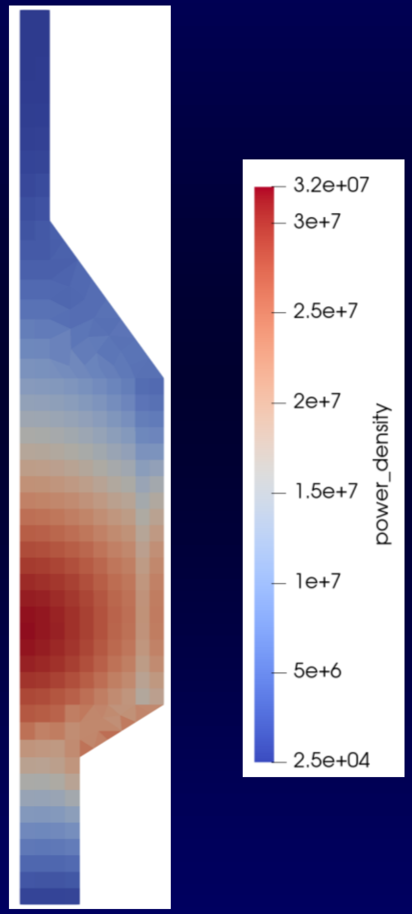


Molten Coolant Salt Reference Plant (PB-FHR)

- Coupled model developed using SAM/Griffin
- Equilibrium core determined with streamline depletion method.
- Simulations conducted for:
  - Steady-state power, temperature, flow distribution
  - Control rod withdrawal event
  - Unprotected loss-of-flow



Fluid Temperature



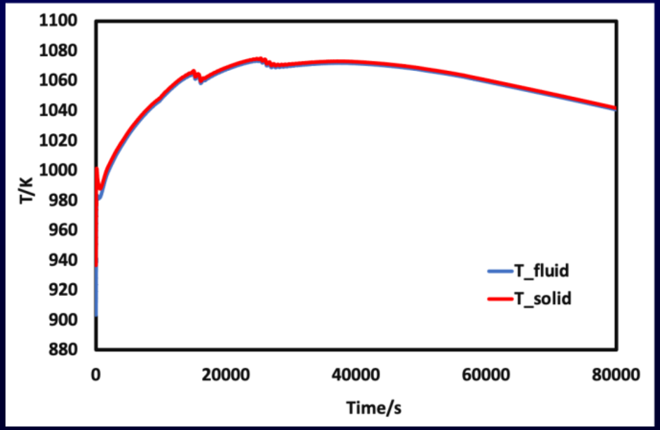
Power Density

Credits: J. Ortensi et al (INL), R. Hu, et al. (ANL)

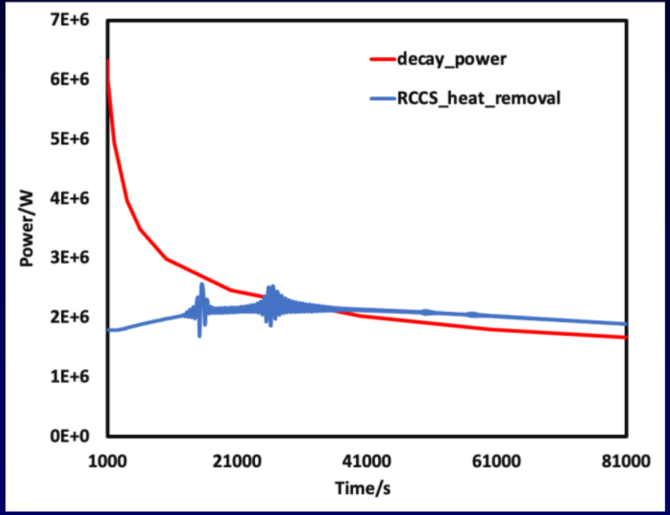
Molten Coolant Salt Reference Plant (PB-FHR)

- Unprotected loss-of-flow scenario
  - Pump coasts down over 75 seconds
  - SCRAM does not occur
  - Fluid diode opens and natural circulation is established between core and downcomer
  - Core heats resulting in decrease in prompt power
  - Decay power removed by RCCS

Improved model coming soon !



Core Solid and Coolant Temperatures



Heat Removal by RCCS

Credit: J. Ortensi, et al (INL)

Molten Fuel Salt Reference Plant

- Molten Salt Reactor Experiment (MSRE) doubles as both a reference plant and for code validation.

ORNL successfully demonstrated key MSR technology at the MSRE

Salt chemistry was well behaved (almost no corrosion)
Nuclear performance closely paralleled predictions
Molten-salts stable under reactor conditions

- 1965 (June) First Criticality
- 1966 (Dec) First Full Power Operation
- 1968 (Oct) First Operation on U-233
- 1969 (Dec) Shutdown
- Design features:
  - 8 MWt
  - Single region core - 33% U-235
- Graphite moderated
- Alloy N vessel and piping
- Achievements
  - First use of U-233 Fuel
  - First use of mixed U/Pu salt fuel
  - On-line refueling
  - >13,000 full power hours

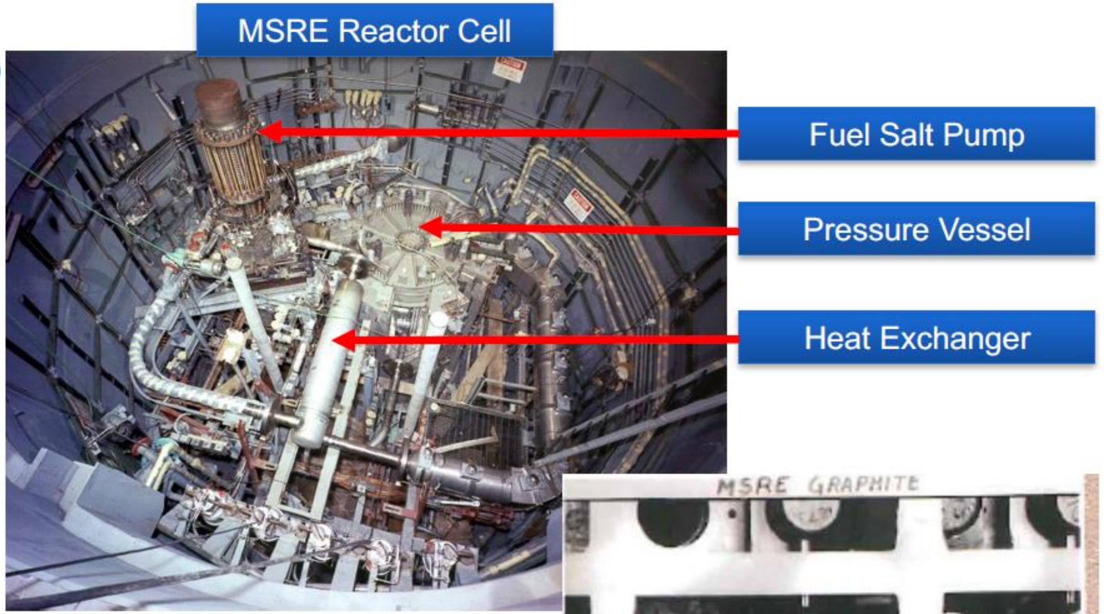
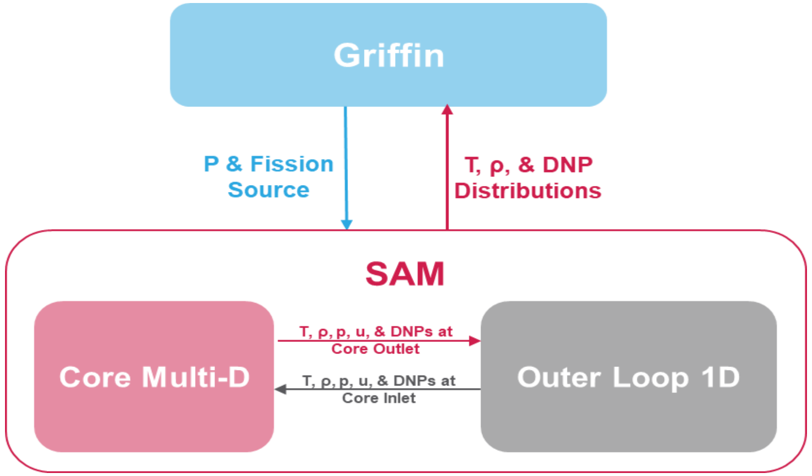
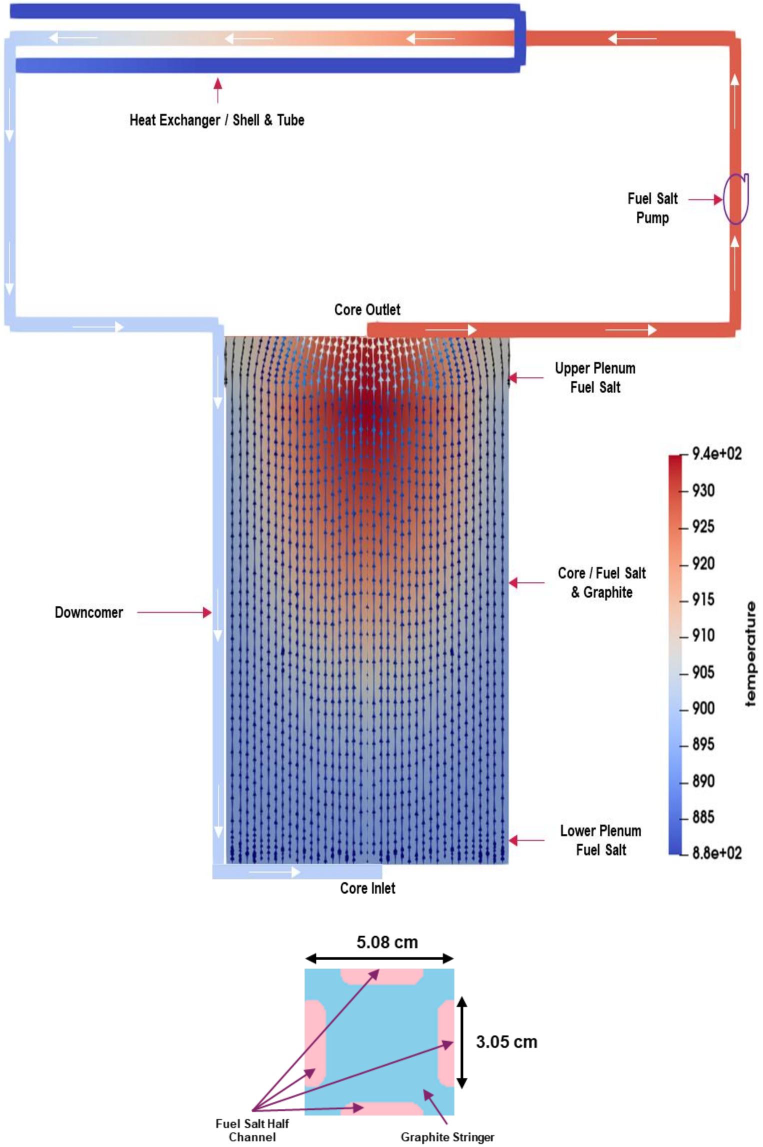


Image shows chemical compatibility of the las with graphite. The shape of this graphite rod is virtually identical in the left image from 1964 before operation and in right image from 1970 after shutdown.

MSRE Multiphysics Model

- **A Multiphysics** model of the **MSRE** was developed in **RZ** geometry including the following components:
  - Neutronics core model of core: Griffin
  - Thermal hydraulics core model: SAM Multi-D
  - Thermal hydraulics outer loop model: SAM 1-D





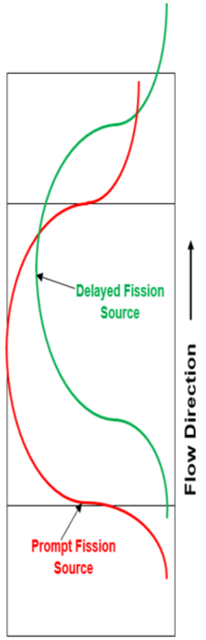
- **Three feedback mechanisms**:
  - **Temperature**: Fuel Salt and Graphite.
  - **Density**: concentration of the salt nuclides due to salt expansion.
  - **Velocity**: delayed neutron precursors distributions in core & outer loop.

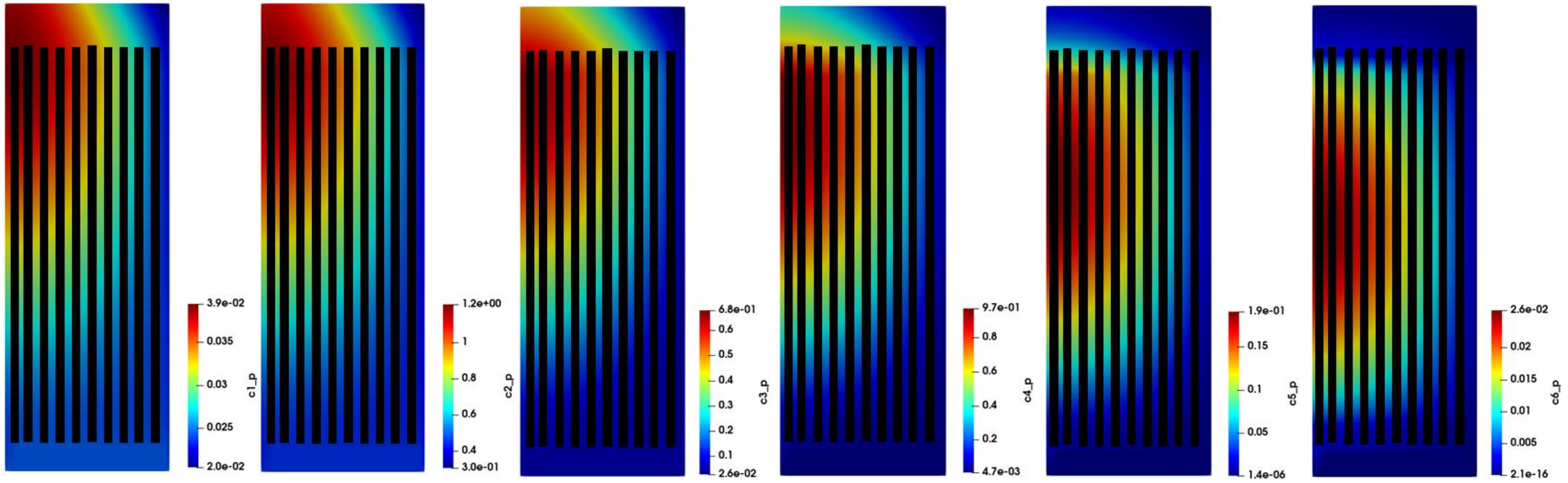
Credits: J. Ortensi et al (INL), R. Hu, et al. (ANL)

Delayed Neutron Precursors Steady State Solution

Long Lived → Short Lived

	Group 1	Group 2	Group 3	Group 4	Group 5	Group 6
	$\lambda_1 = 0.013\ s^{-1}$	$\lambda_2 = 0.033\ s^{-1}$	$\lambda_3 = 0.121\ s^{-1}$	$\lambda_4 = 0.303\ s^{-1}$	$\lambda_5 = 0.849\ s^{-1}$	$\lambda_6 = 2.853\ s^{-1}$
	$T_{1/2\_1} = 51.976\ s$	$T_{1/2\_2} = 21.172\ s$	$T_{1/2\_3} = 5.739\ s$	$T_{1/2\_4} = 2.289\ s$	$T_{1/2\_5} = 0.816\ s$	$T_{1/2\_6} = 0.243\ s$





Flowing	$\beta_1 = 0.000006$	$\beta_2 = 0.000374$	$\beta_3 = 0.000535$	$\beta_4 = 0.001846$	$\beta_5 = 0.001048$	$\beta_6 = 0.000481$
Stationary	$\beta_1 = 0.000237$	$\beta_2 = 0.001218$	$\beta_3 = 0.001163$	$\beta_4 = 0.002592$	$\beta_5 = 0.001068$	$\beta_6 = 0.000447$
Losses	$\beta_1 = 0.000231$	$\beta_2 = 0.000844$	$\beta_3 = 0.000628$	$\beta_4 = 0.000746$	$\beta_5 = 0.000020$	$\beta_6 = 0.000034$

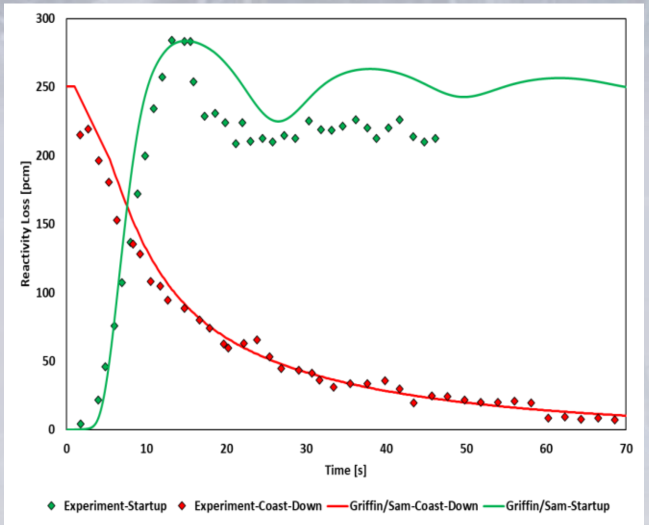
Calculated total reactivity losses due to fuel salt flow is 240 pcm.

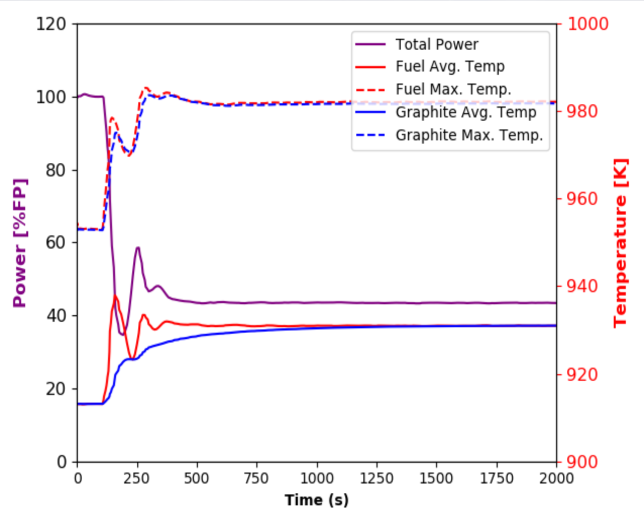
Credits: J. Ortensi et al (INL), R. Hu, et al. (ANL)

MSRE Simulations

- Simulations performed for:
  - Pump start-up
  - Pump coastdown
  - Unprotected Loss-of-Flow (ULOF) at zero power
  - Unprotected Loss-of-Flow (ULOF) at full power

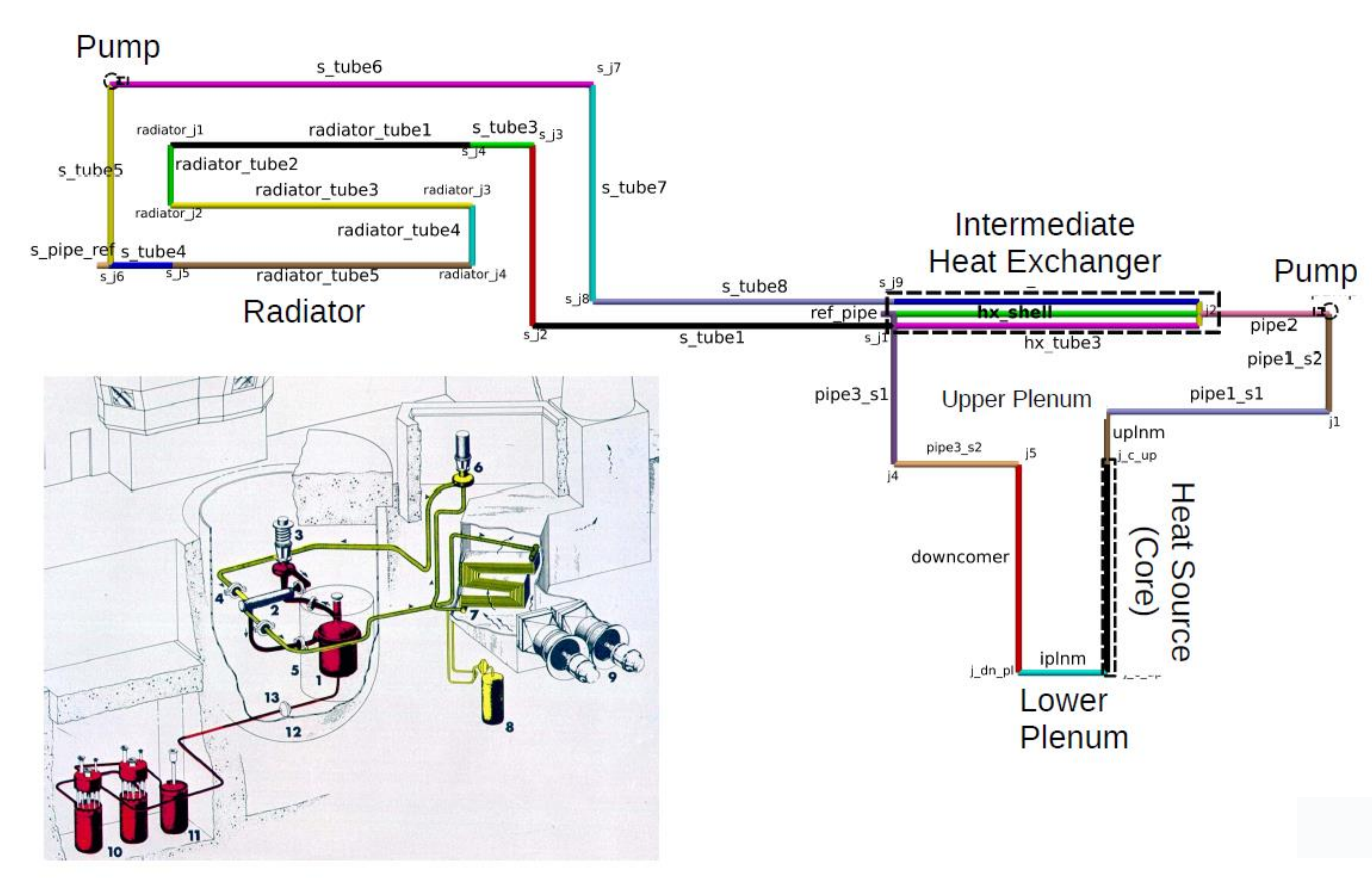
- Sensitivity Study on natural circulation (1D)





Credits: J. Ortensi et al (INL), R. Hu, et al. (ANL)

MSRE 1D Model

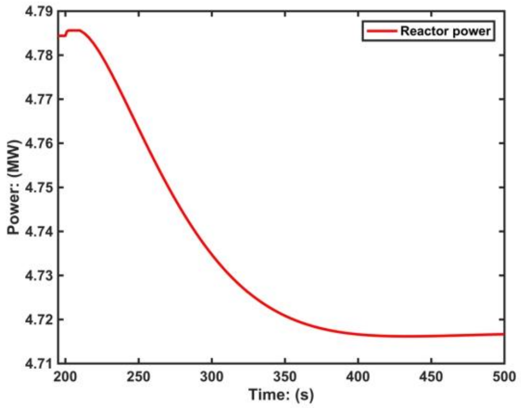


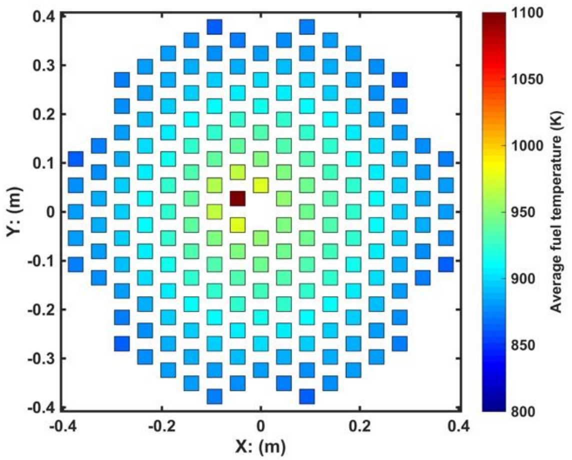
Heat Pipe Cooled Microreactors

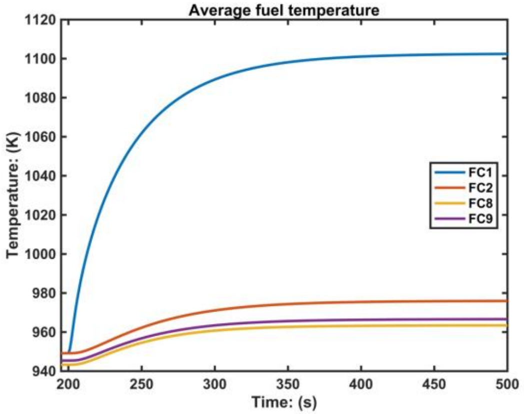
- Reference Models developed for two types of heat pipe cooled microreactors.
  - "Modified SPR-A" which is a modified version of the "Special Purpose Reactor" as a heat pipe cooled fast reactor with metallic fuel. Orientation is vertical.
  - "eVinci-like" which is based on public information on a heat pipe cooled thermal reactor with TRISO fuel within a graphite monolith. Orientation is horizontal.

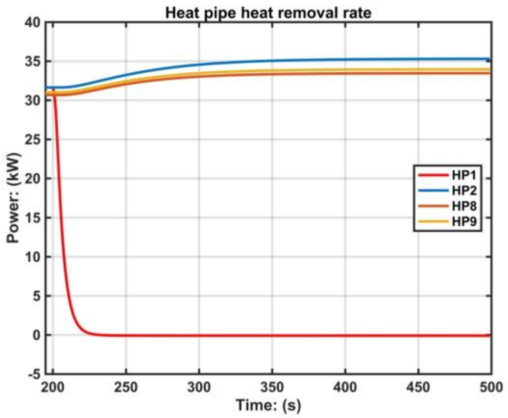
SINGLE HEAT PIPE FAILURE

- 1 of central heat pipe is assumed to be failed (HP1)
- The reactor power re-stabilizes to about 4.718 MW after 300 s of the transient
- Fuel temperature of the failed cell increases about 150 K
- Fuel temperature increase in the neighboring cells is limited, ~ 20 K
- Heat removal rate of neighboring heat pipes increases





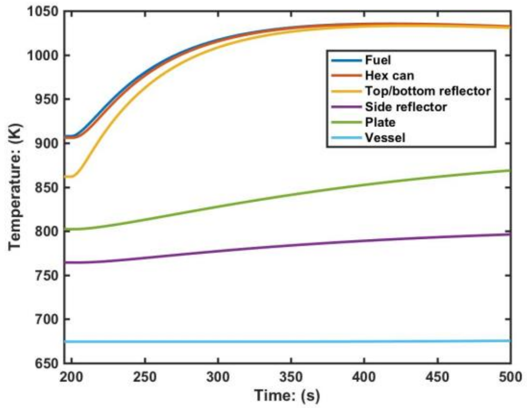




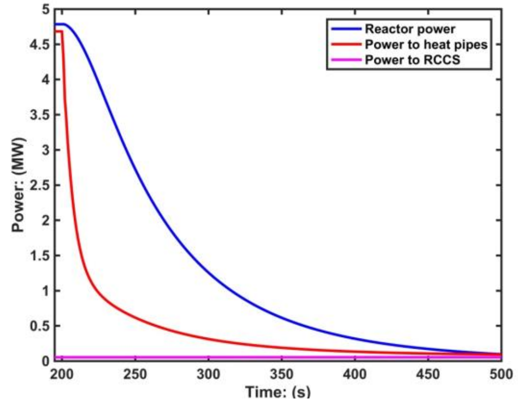
Credit: J. Ortensi, et al (INL), J. Kelly (NRC)

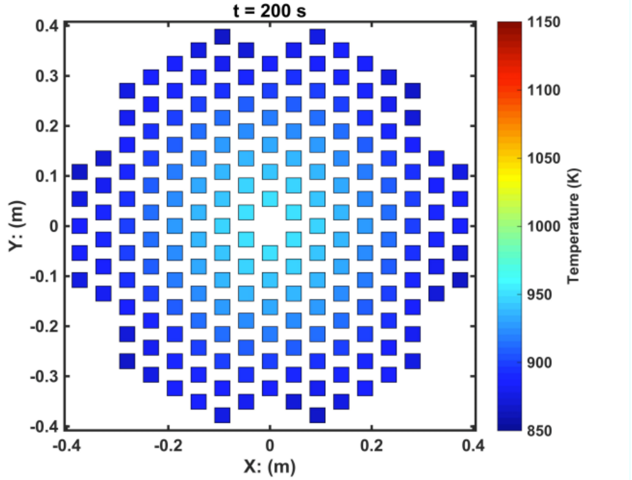
LOSS OF HEAT SINK

- Heat pipe heat removal rate drops quickly to a lower level
  - Flow rate drops to 0.1% of steady-state value
  - Slow decrease due to the thermal inertial of the heat pipes
- Reactor power drops quickly due to the strong negative reactivity feedback
- Decay power was not considered yet in the reactor physics model



Average solid temperature





Fuel average temperature

Credits: J. Ortensi et al (INL), J. Kelly (NRC)

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Advanced Reactors
Three-Phased Approach for Confirmatory Models

Stage 1 – Generic Readiness for a Reactor Technology

Code infrastructure development, reference plant model/source term demonstration, generic models that benefit all non-LWR designs (IAP Strategy 2 Volumes)

Stage 2 – Readiness for a Specific Application

Model build of a preapplication based design)

Stage 3 – Model build, Analysis, and Review of a specific application under licensing review

Conduct confirmatory analysis, generation of RAIs, and input to SER

Next Steps

– Improve upon existing Reference Models:
  - Multi-dimensional core models for 3D asymmetric events
  - Improved secondary loop models and more accurate RCCS.
  - More accurate core power distributions

– Additional validation
  - Address assessment gaps
  - Work with DOE and applicants on database insufficiencies

– Further prepare for applicant submittals based on improved public and proprietary information.

Summary & Conclusions

RES

– Code development and preparation for independent analysis of non-LWRs with the BlueCRAB system codes is well underway and making significant progress.

– Reference plant models are available and being tested for each “near term” applicant design.

– Verification and Validation (V&V) has been documented in a separate report and can be used to identify weaknesses in the available database and assessment.

– BlueCRAB is “tentatively ready” for independent analysis of non-LWRs and is available to support the licensing and evaluation process.


Questions?

Status Update on Computer Code and Model Development for Non-Light-Water Reactors

April 3, 2024

Fuel Performance Analysis

James Corson, Ph.D.

Senior Reactor System Engineer

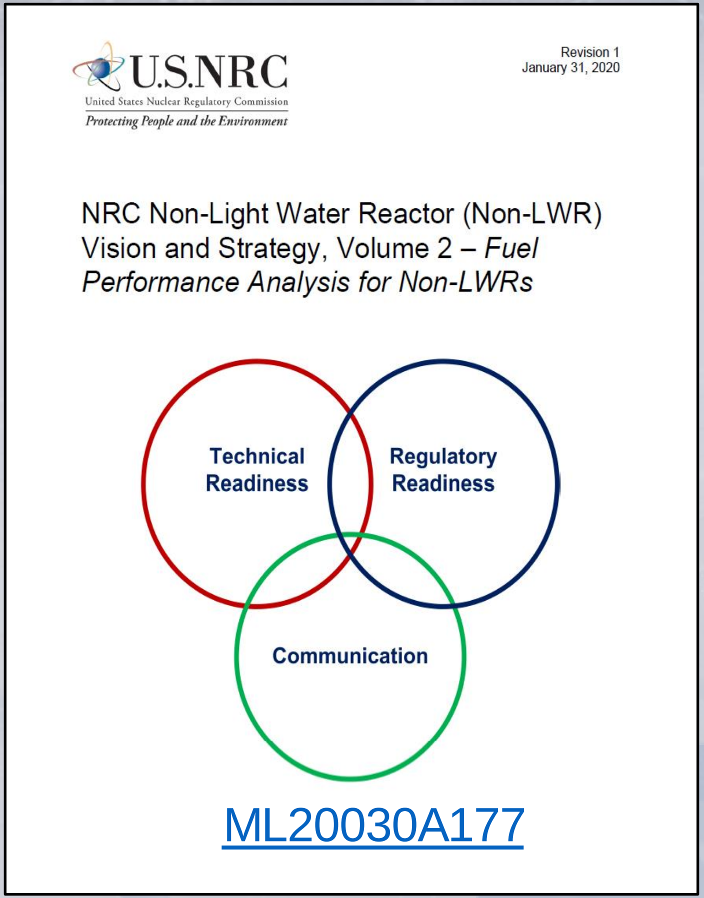
Division of Systems Analysis

Office of Nuclear Regulatory Research

Fuel Performance Analysis Objectives

RES

- Understand thermal-mechanical nuclear fuel performance during normal operations, anticipated operational occurrences, and accident conditions
  - ➢ Provide insights for regulatory guidance
  - ➢ Support topical report reviews
- Ensure tool & model readiness for licensing non-LWRs
  - ✓ Develop necessary modeling capabilities in FAST
  - ✓ Perform assessments against available experimental data


U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Revision 1
January 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 2 – *Fuel Performance Analysis for Non-LWRs*

ML20030A177

Fuel Performance Analysis FAST Code

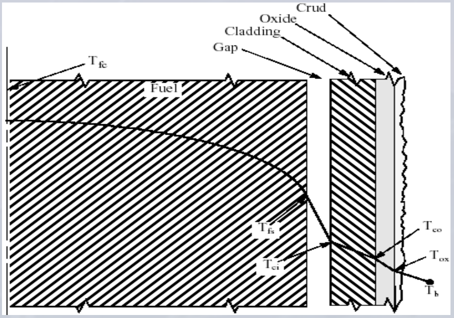
RES

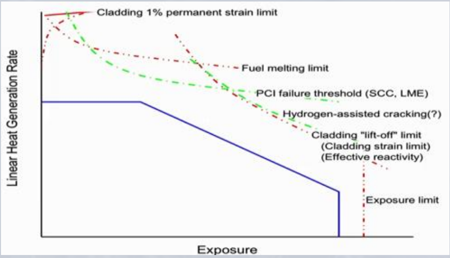
What Is It?

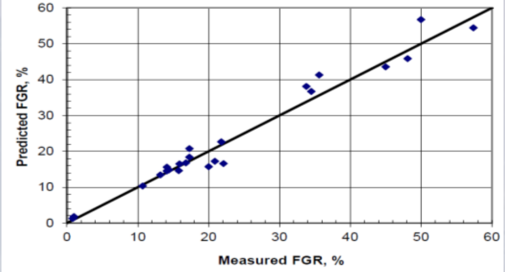
FAST (Fuel Analysis under Steady-State & Transients) calculates the thermal-mechanical response of nuclear fuel under steady-state and accident conditions.

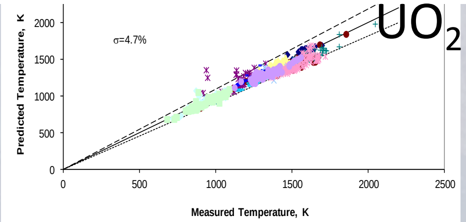
How Is It Used?

FAST is used to support licensing reviews by assessing specified acceptable fuel design limits, evaluating vendor fuel codes and methods, and providing initial conditions for design-basis accident analysis. It is also used to perform spent fuel analyses.





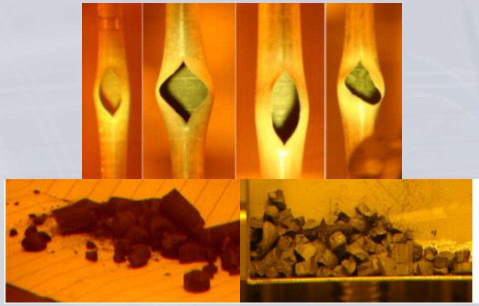




Halden Reactor Project	Studsvik Cladding Integrity Project (Phase I - V)	Second Framework for Irradiation Experiments (FIDES-II)	Cabri International Project	OECD/NEA's QUENCH-ATF Program	Lead Test Assemblies & Lead Test Rods Programs	DOE's Advanced Gas Reactor Program (AGR)	DOE's Sibling Rod Program
1980s-Present	**2004-Present**	**2021-2024**	**2000-Present**	**2021-Present**	**Ongoing**	**2002-Present**	**2016-Present**

Who Uses It?

FAST is used by more than 75 domestic and international organizations, including other regulatory bodies, technical scientific organizations, and utilities, for safety and core reload applications.

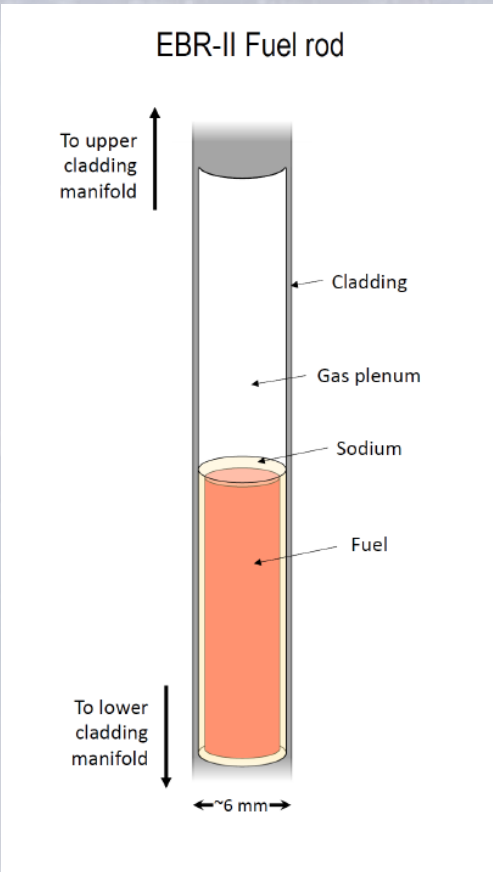


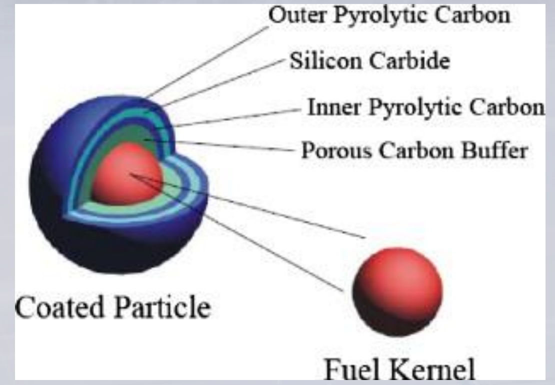
How Has It Been Assessed?

FAST is built on more than 30 years of assessment stemming from the FRAPCON/FRAPTRAN codes, as well as experience with fuel vendor codes and data. It offers more than 200 assessment cases that cover the UO_2/zirconium fuel system, and new cases added for metallic fuels.

Fuel Performance Analysis Approach for Metallic and TRISO Fuel Forms

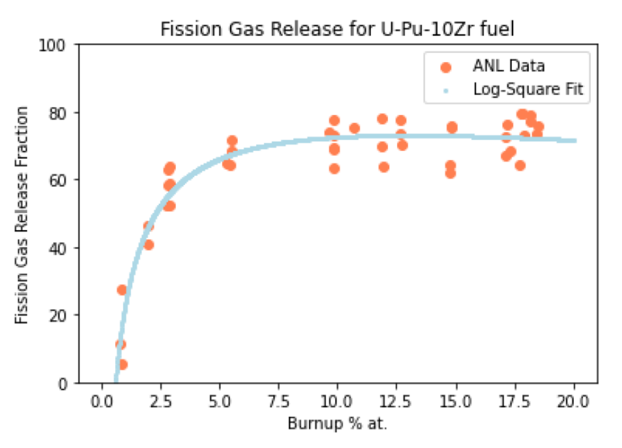
1. Update FAST with relevant models for metallic (U-xPu-10Zr) and TRISO fuels
2. Assess the code against relevant experimental data
   a. EBR-II and FFTF for metallic fuel
   b. AGR for TRISO





Fuel Performance Analysis Metal Fuel Models in FAST

- Existing U-10Zr fuel, HT-9 cladding models are empirical, based primarily on EBR-II experience
  - Anisotropic fuel swelling fitted to experimental data
  - Fission gas release fitted to experimental data
- Future work needed for fuel failure models and to extend beyond the existing database
  - Fuel clad chemical interaction (FCCI) model
  - Cladding overpressure failure models
  - More mechanistic swelling and fission gas release models



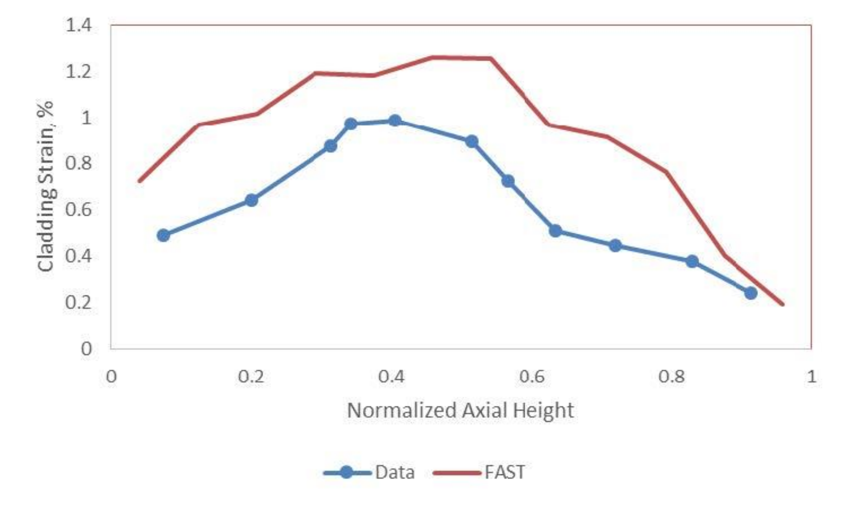
FGR data from Pahl et al., *JNM* 188 (1992) 3

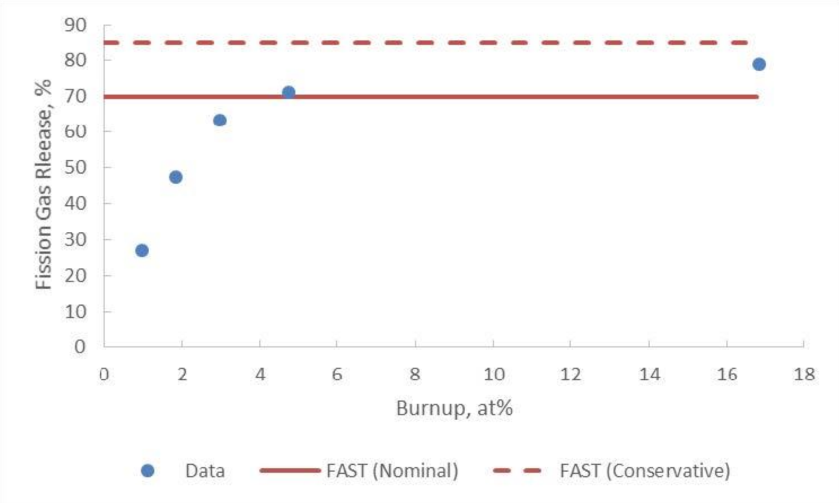
Fuel	Cladding	Depleted Zone, µm	Burnup, at%	Temperature, ºC
U-15Pu-9Zr	304L 316	140 30	5 5	650 650
U-9Pu-10Zr	D9 316	100 70	17 13	580 580
U-10Zr	D9	20	17	580
U-10Zr	HT9	100	5	650
U-19Pu-10Zr	HT9	45	12	600

FCCI data from Hofman et al., *Progress in Nuclear Energy* 31 (1997) 83

Fuel Performance Analysis
Preliminary FAST Assessment (Metal Fuels)

- Initial assessments performed in 2018
    - Included constant swelling and FGR rates
    - Updated assessment using new models in progress
- Improved models can reduce uncertainties
    - Currently updating our earlier assessments

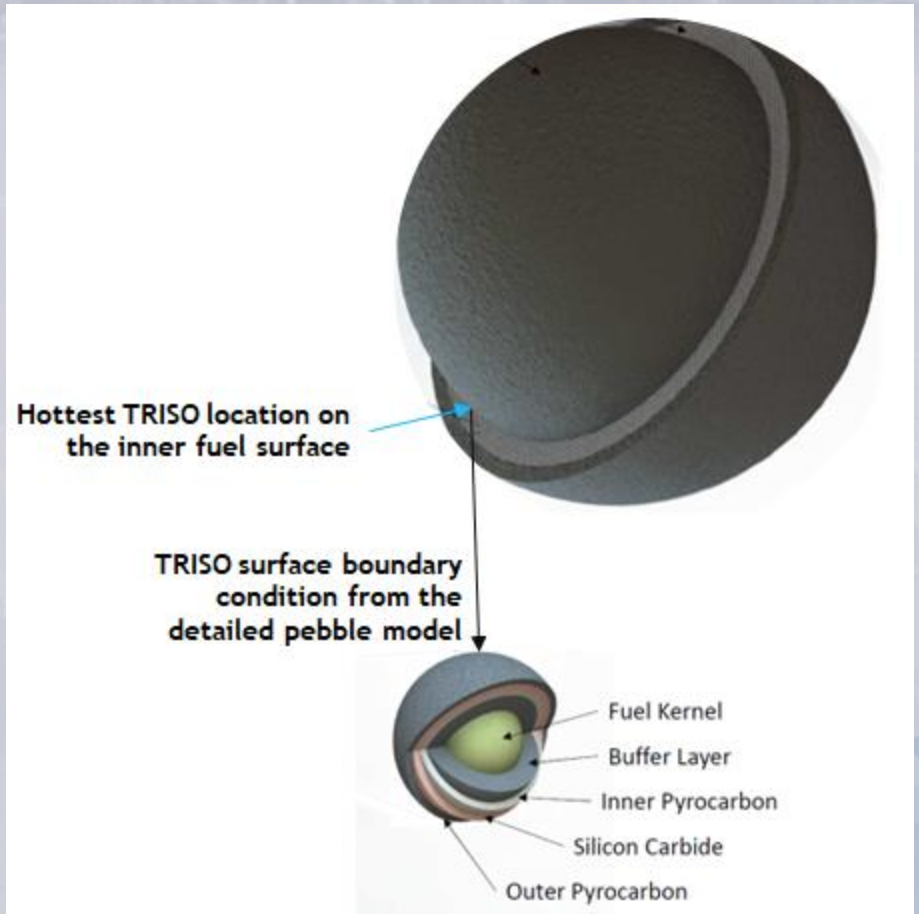




Geelhood & Porter, Top Fuel 2018

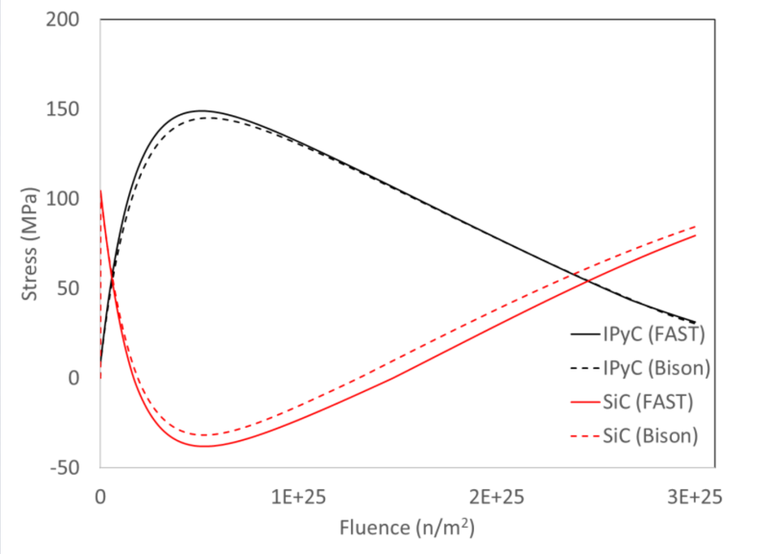
Fuel Performance Analysis FAST-TRISO

- New Standalone 1D code for TRISO fuel performance
  - Focuses on uranium oxycarbide (UCO) kernels surrounded by buffer, inner pyrocarbon (IPyC), silicon carbide (SiC), and outer pyrocarbon (OPyC) layers
- Latest release includes the following capabilities
  - Heat transfer from the kernel to the particle surface
  - Stresses in PyC and SiC layers
  - Fission product transport from the kernel through the layers
  - Monte Carlo analysis for layer failure probabilities

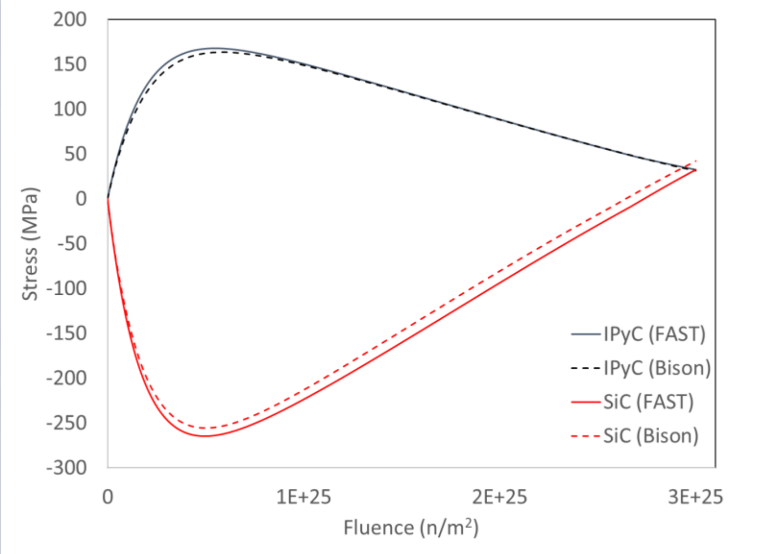


Fuel Performance Analysis
Ongoing TRISO Work

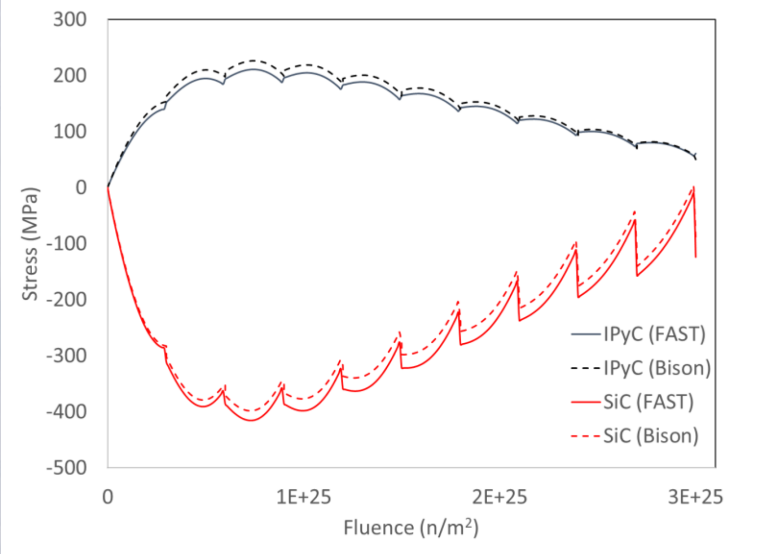
- Code development
  - Mechanical model recently extended to include PyC swelling and creep
  - Currently developing correlations for stress concentrations due to PyC cracking and debonding and aspherical particles (using Abaqus)
- Code assessment
  - Results in good agreement with CRP-6 fuel performance cases 1-8 in IAEA-TECDOC-1674
  - Work comparing to AGR fission product release and failure data ongoing



CRP-6 Case 4d



CRP-6 Case 6



CRP-6 Case 8

Fuel Performance Analysis Summary

RES

- NRC fuel performance codes are ready for confirmatory analysis of U(Pu)-10Zr and UCO TRISO
  - More assessments against EBR-II (metallic fuel) and AGR (TRISO) would be beneficial
  - Longer-term goal is to add more mechanistic models informed by data from DOE's Advanced Fuels Campaign and NEAMS code development efforts
- Code development efforts have significantly built staff expertise on advanced reactor fuel behavior

Status Update on Computer Code and Model Development for Non-Light-Water Reactors

April 3, 2024

Severe Accident Analysis

Shawn Campbell, Ph.D.
Reactor System Engineer

Lucas Kyriazidis
Reactor System Engineer

Andrew Bielen, Ph.D.
Senior Reactor System Engineer

Division of Systems Analysis
Office of Nuclear Regulatory Research

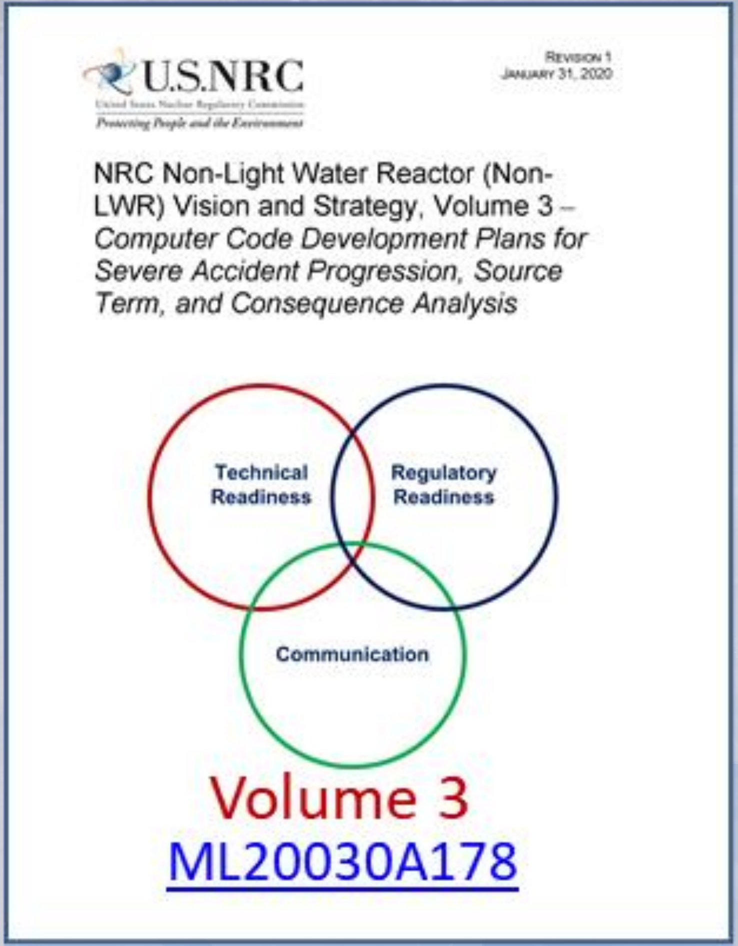
Severe Accident Analysis Objectives

RES

- Understand severe accident progression in non-LWRs
  - Provide insights for regulatory guidance
  - Build staff knowledge and expertise in modeling non-LWRs

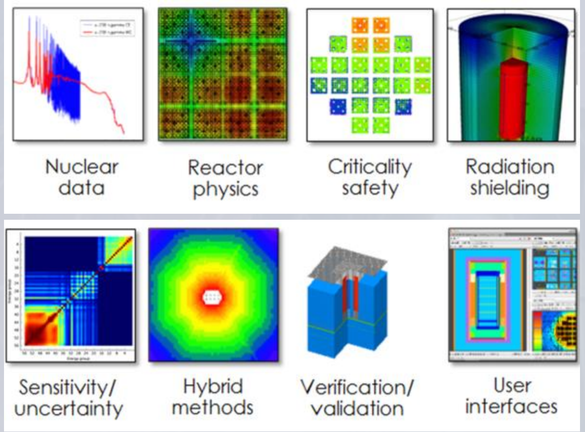
- Facilitate dialogue on staff’s approach for source term

- Ensure tool & model readiness for licensing non-LWRs
  - Develop necessary modeling capabilities in SCALE & MELCOR
  - Identify accident characteristics and uncertainties affecting source term

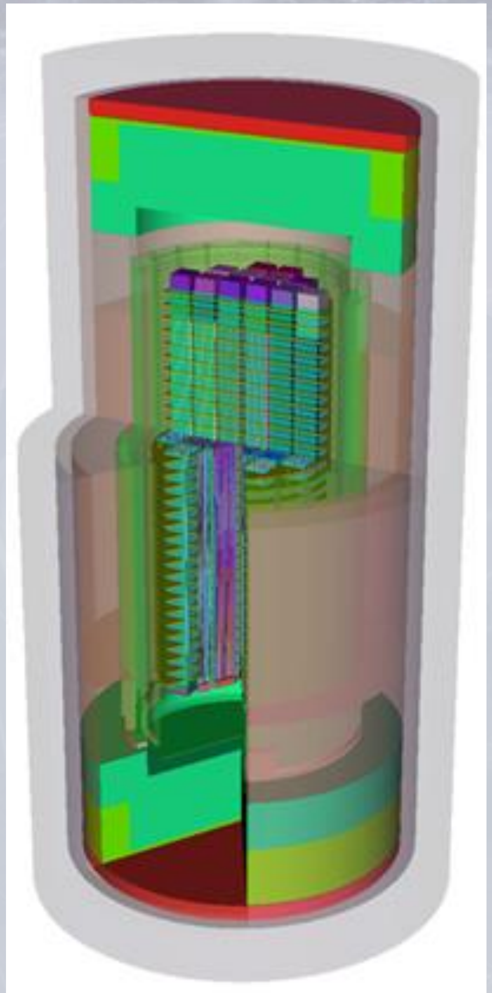


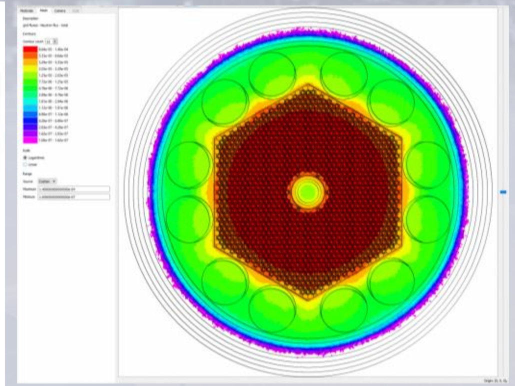
Severe Accident Analysis SCALE Code

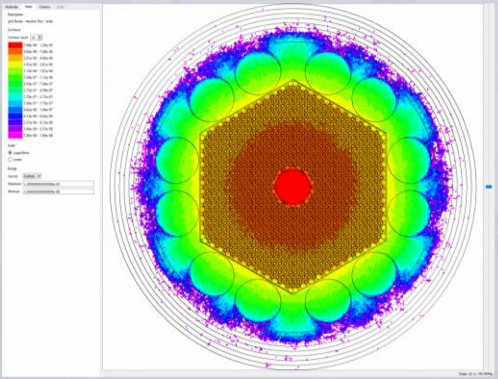
What Is It?
The SCALE code system is a modeling and simulation suite for nuclear safety analysis and design. It is a modernized code with a long history of application in the regulatory process.



How Is It Used?
SCALE is used to support licensing activities (e.g., analysis of spent fuel pool criticality, generating reactor physics and decay heat parameters for design-basis accident analysis, and review of consolidated interim storage facilities, burnup credit).







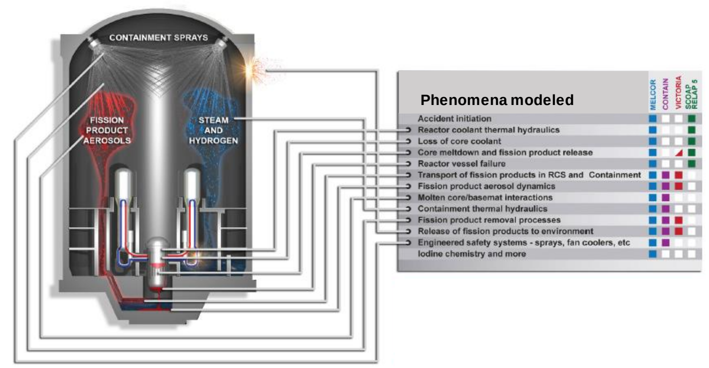
Who Uses It?
SCALE is used by the NRC and in 61 countries (about 11,000 users and 33 regulatory bodies).

How Has It Been Assessed?
SCALE has been validated against numerous critical experiments that cover a range of fuel and moderator materials and geometries, and against measured PWR and BWR spent fuel isotopic composition and decay heat measurements.

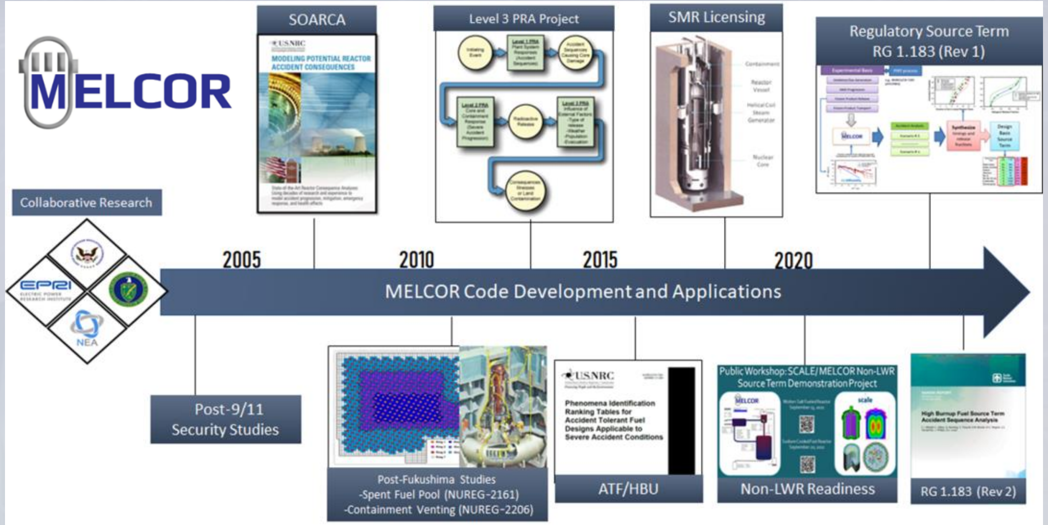
Severe Accident Analysis MELCOR Code

RES

What Is It?
MELCOR is an engineering-level code that simulates the response of the reactor core, primary coolant system, containment, and surrounding buildings to a severe accident.



How Is It Used?
MELCOR is used to support severe accident and source term activities at the NRC, including the development of regulatory source terms; support for probabilistic risk assessment models and site risk studies; containment analysis; and forensic investigations of the Fukushima accident.



Phébus-Fission Products & Source Term Program	Behavior of Iodine Project (BIP)	Experimental Program for Iodine Chemistry Under Radiation (EPICUR)	Source Term Evaluation and Mitigation (STEM) Project	Benchmark Study of the Accident at Fukushima (BSAF) Project	Management and Uncertainties of Severe Accidents (MUSA)	Experiments on Source Term for delayed Releases (ESTER) Reduction of Severe Accident Uncertainties (ROSAU)	Thermodynamic Characterization Of Fuel debris and Fission (TCOFF-2)	Fukushima Accident Information Collection & Evaluation (FACE)
1988-2010	2006-2019	2005-2016	2011-2019	2013-2018	2019-2023	2020-2024	2022-2024	2023-2026

Who Uses It?
MELCOR is used by domestic universities and national laboratories and around 30 international organizations. It is distributed as part of the NRC's Cooperative Severe Accident Research Program (CSARP).

How Has It Been Assessed?
MELCOR has been validated against numerous international standard problems, benchmarks, separate effects (e.g., VERCORS) and integral experiments (e.g., Phebus FPT), and reactor accidents (e.g., TMI-2, Fukushima).

Severe Accident Analysis Approach

1. Build representative SCALE core models and MELCOR full-plant models

2. Select scenarios that demonstrate code capabilities for key phenomena

3. Perform simulations
   - SCALE - generate decay heat, core radionuclide inventory, and reactivity feedbacks
   - MELCOR - model accident progression, plant response, and source term

Severe Accident Analysis Project Scope

- Five Types of Non-LWRs Analyzed for Source Term Demonstration

- *2021*
  - Heat Pipe Reactor – INL Design A
  - High-Temperature Gas-cooled Pebble-bed Reactor – PBMR-400
  - Molten-salt-cooled Pebble-bed Reactor – UCB Mark 1

- *2022*
  - Molten-salt-fueled Reactor – MSRE
  - Sodium-cooled Fast Reactor – ABTR

SCALE/MELCOR non-LWR source term demonstration project	

• Heat-pipe reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	June 29, 2021

• High-temperature gas-cooled reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	July 20, 2021

• Fluoride-salt-cooled high-temperature reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 14, 2021

• Molten-salt-fueled reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 13, 2022

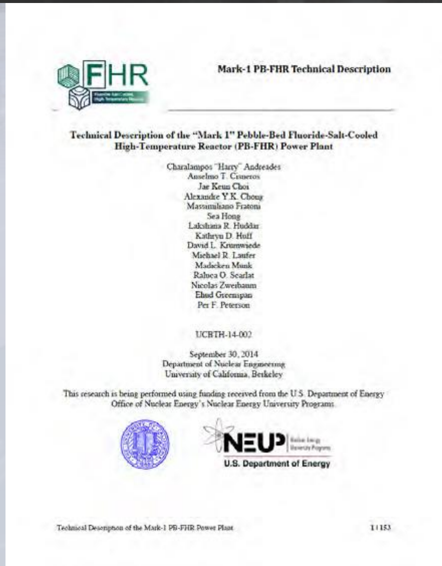
• Sodium-cooled fast reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 20, 2022

Public workshop videos, slides, reports at advanced reactor source term webpage

SCALE input models available here.

MELCOR input models available upon request.

Severe Accident Analysis
Molten-salt-cooled Pebble-bed Rx – UCB Mark 1


FHR

Mark-1 PB-FHR Technical Description

Technical Description of the "Mark 1" Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant

Charalampos "Harry" Andreades
Anselmo T. Cisneros
Jae Keun Choi
Alexandre Y.K. Chong
Massimiliano Fratoni
Sea Hong
Lakshana R. Huddar
Kathryn D. Huff
David L. Krumwiede
Michael R. Laufer
Madicken Munk
Raluca O. Scarlat
Nicolas Zweibaum
Ehud Greenspan
Per F. Peterson

UCBTH-14-002

September 30, 2014
Department of Nuclear Engineering
University of California, Berkeley

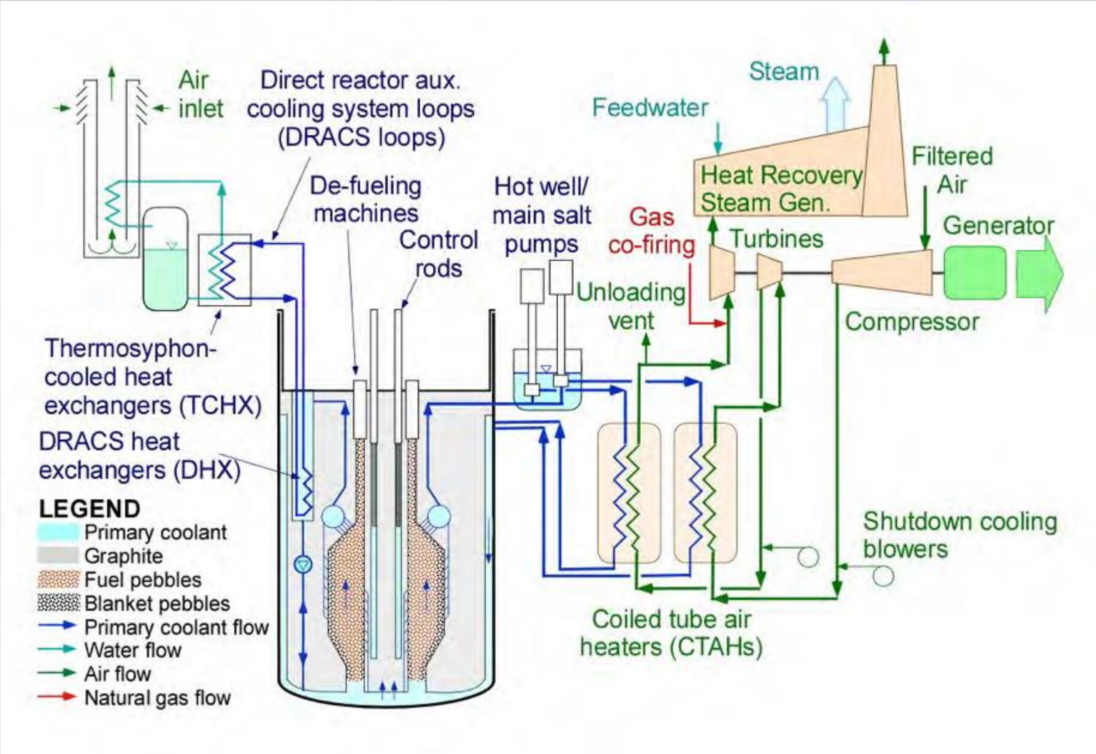
This research is being performed using funding received from the U.S. Department of Energy Office of Nuclear Energy's Nuclear Energy University Programs

NEUP
U.S. Department of Energy

Technical Description of the Mark-1 PB-FHR Power Plant 1 | 153

<u>Reactor Characteristics</u>

- 236 MWth reactor
- Atmospheric pressures
- Flibe cooled
- Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling
- Direct Reactor Auxiliary Cooling System (DRACS)
  - 3 trains –2.36 MW/train
  - Each train has 4 loops in series
  - Primary coolant circulates to DRACS heat exchanger
  - Molten-salt loop circulates to the thermosyphon-cooled heat exchangers (TCHX)
  - Water circulates adjacent to the secondary salt tube loop in the TCHX



<u>Accidents Modeled</u>

- ATWS –Anticipated transient without SCRAM
- SBO –Station blackout
- LOCA –Loss-of-coolant accident

Severe Accident Analysis
Molten-salt-cooled Pebble-bed Rx – UCB Mark 1

RES

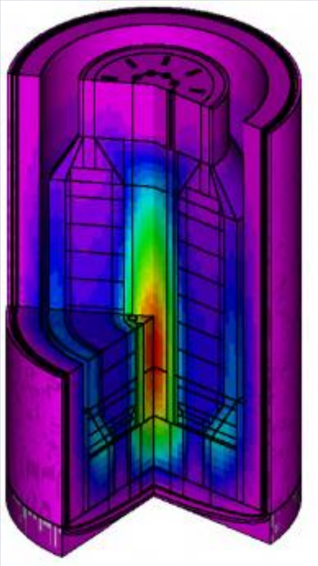
scale

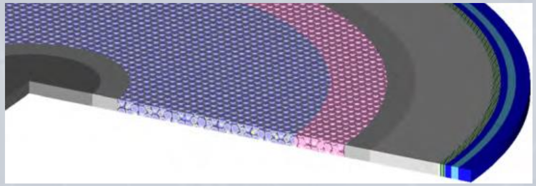
Code Improvements

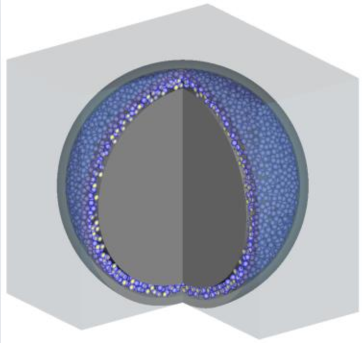
MELCOR

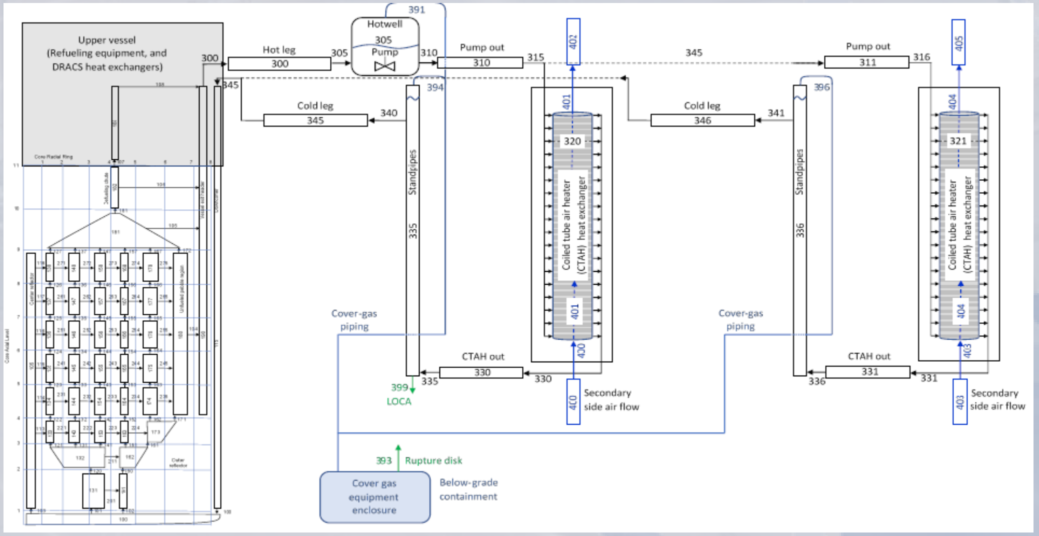
- New interface for rapid depletion of TRISO fuel for low computational costs (*increased efficiencies for performing wide array of sensitivity studies)*
- Developed workflow for pebble-bed reactor equilibrium core generation using SCALE's efficient multigroup treatment for double heterogeneous systems

- Added a generic equation of state utility for thermal hydraulic analysis in advanced reactor working fluids
- Fission product transport and retention models added for molten salts
- Improved fission product release models for TRISO
- Point-kinetic enhancements for reactivity insertion









Severe Accident Analysis
Molten-salt-cooled Pebble-bed Rx – UCB Mark 1

ATWS

- Fuel heat-up was limited by reactivity feedback and the passive decay heat removal system

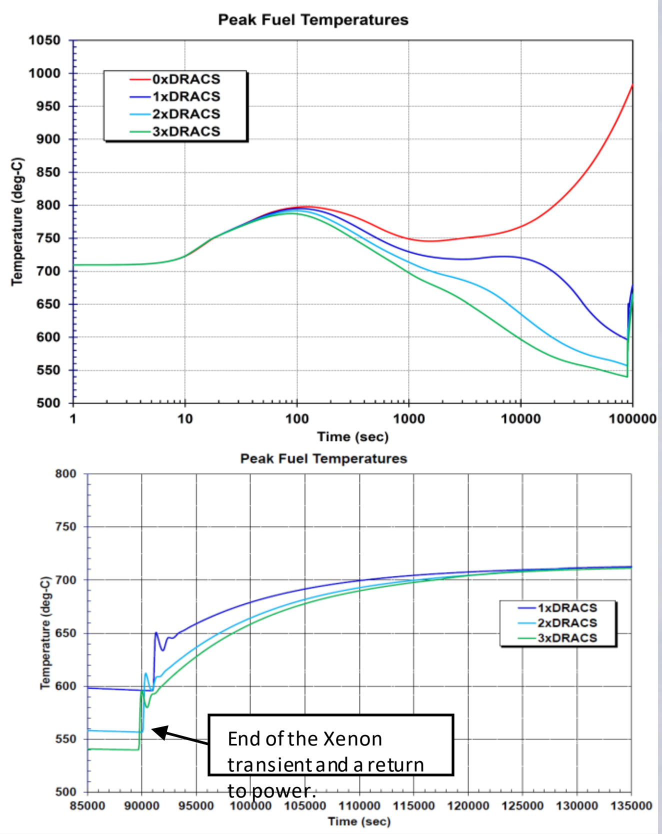
SBO

- With failure of the passive decay heat removal system, coolant boiling occurred over the course of several days

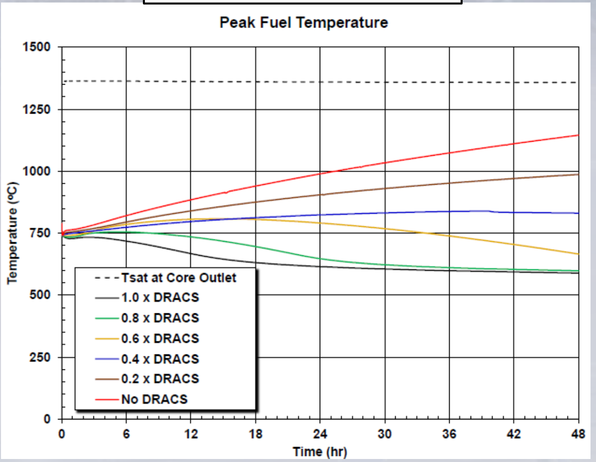
LOCA

- With one train of decay removal system operating, coolant boiling was possibly averted.
- With failure of the passive decay heat removal system, fuel damage occurred.

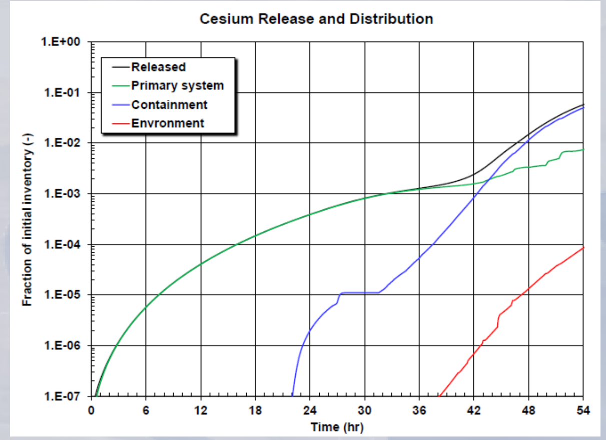
ATWS with variable DRACS



SBO



LOCA

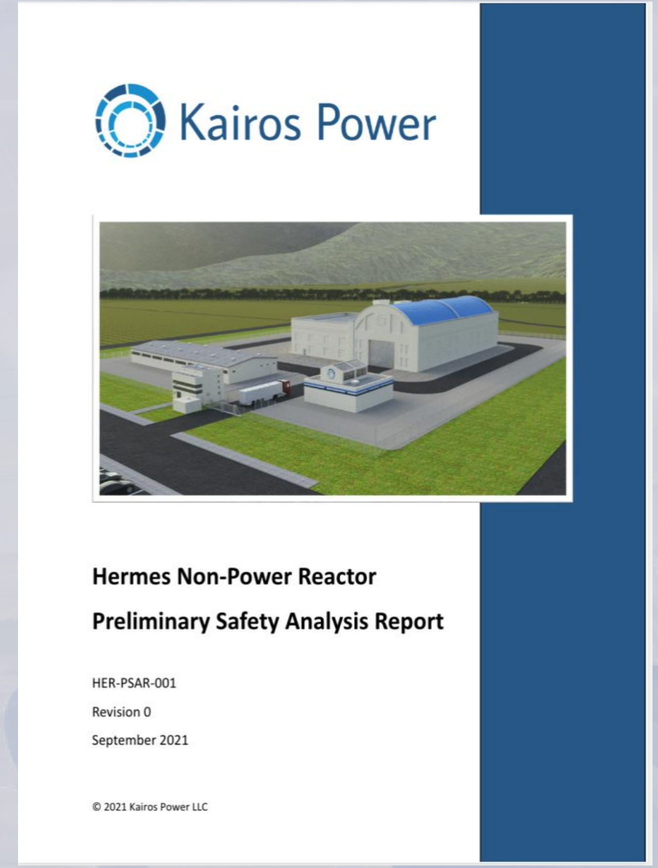


Severe Accident Analysis
Hermes I Construction Permit Application

- On September 29, 2021, Kairos Power, LLC (KP) submitted a construction permit application to the NRC, requesting approval for their Hermes 35 MWth, non-power reactor facility.

- Leverage the UCB-Mark 1 FHR plant model to support Hermes analysis (January-March 2022). Scope was limited to design-basis events (i.e., no fuel uncovery).

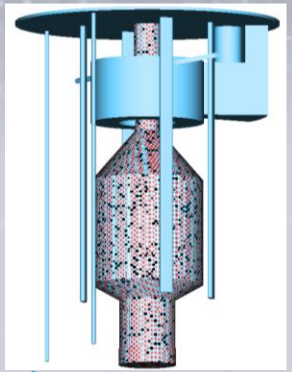
- Provided NRR with SCALE and MELCOR analyses that supported their review looking at:
  - reactor heat-up scenario (e.g., loss of forced circulation),
  - insertion of excess reactivity scenario (e.g., accidental control rod withdrawal)



Severe Accident Analysis
Hermes I: SCALE Model

- Multigroup Monte Carlo transport using SCALE/KENO-VI, fuel isotopics calculated with SCALE/ORIGEN
- Random pebble geometry approximated by regular lattice
- Equilibrium fuel isotopics generated iteratively via 2D slice models with SCALE/TRITON
- Axially-dependent fuel isotopics inserted into 3D core model for reactivity and power shape evaluations
- Does not currently include shutdown (in-bed) elements – on list for further development

Relative Power	Kairos PSAR	SCALE
Axial (-)	1.2	1.19
Radial (-)	1.2	1.76
Peak Pebble (-)	1.8	2.09



Blue: FLiBe
Red: Fuel Pebble
Black: Moderator Pebble

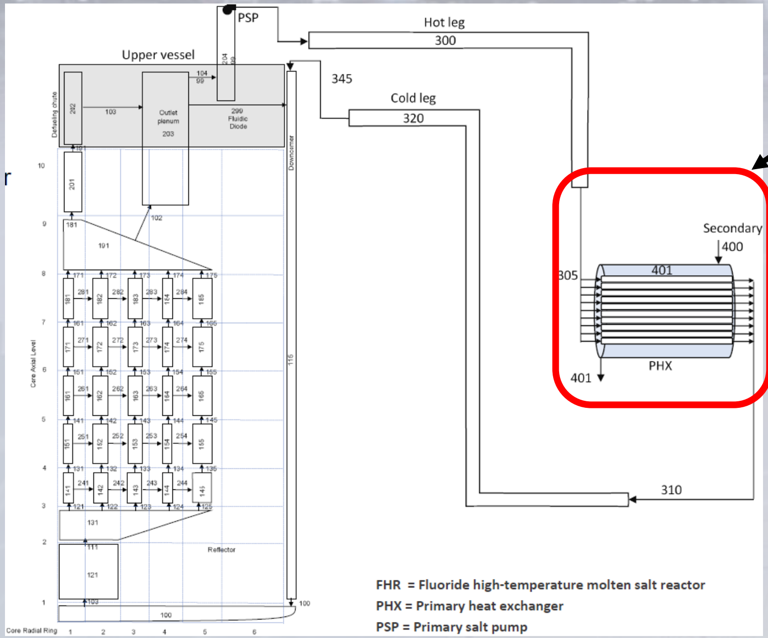
Parameter	Kairos PSAR	SCALE*
Fuel Doppler (pcm/K)†	-4.1	-4.30 ± 0.27
Moderator (pcm/K)†	-0.4	-0.47 ± 0.13
Coolant (pcm/K)†	-1.6	-1.62 ± 0.02
Void (pcm/% void, @3% void)	-53	-46.6 ± 4.0
Reflector (pcm/K)†	+2.0	+1.92 ± 0.23
β_{eff} (pcm)	605	576 ± 10

* - includes Monte Carlo uncertainty
† - calculated assuming temperature distributions provided by MELCOR

Severe Accident Analysis
Hermes I: MELCOR Model

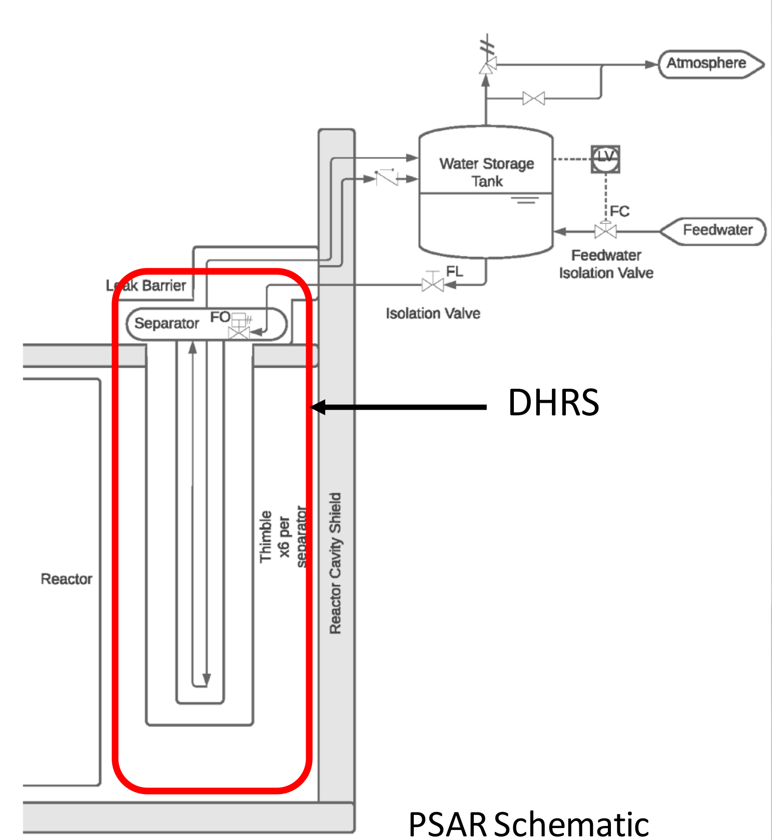
- Model focuses on primary system
  - Secondary system and DHRS represented via boundary conditions
  - Necessary given lack of detailed design info

- DHRS model
  - Water (constant boundary condition at 100°C)
  - Water to DHRS evaporator tube wall uses boiling heat transfer coefficient
  - Thermal resistance between evaporator tube to thimble casing



Secondary system

Figure 6.3-1: Functional Diagram of the DHRS



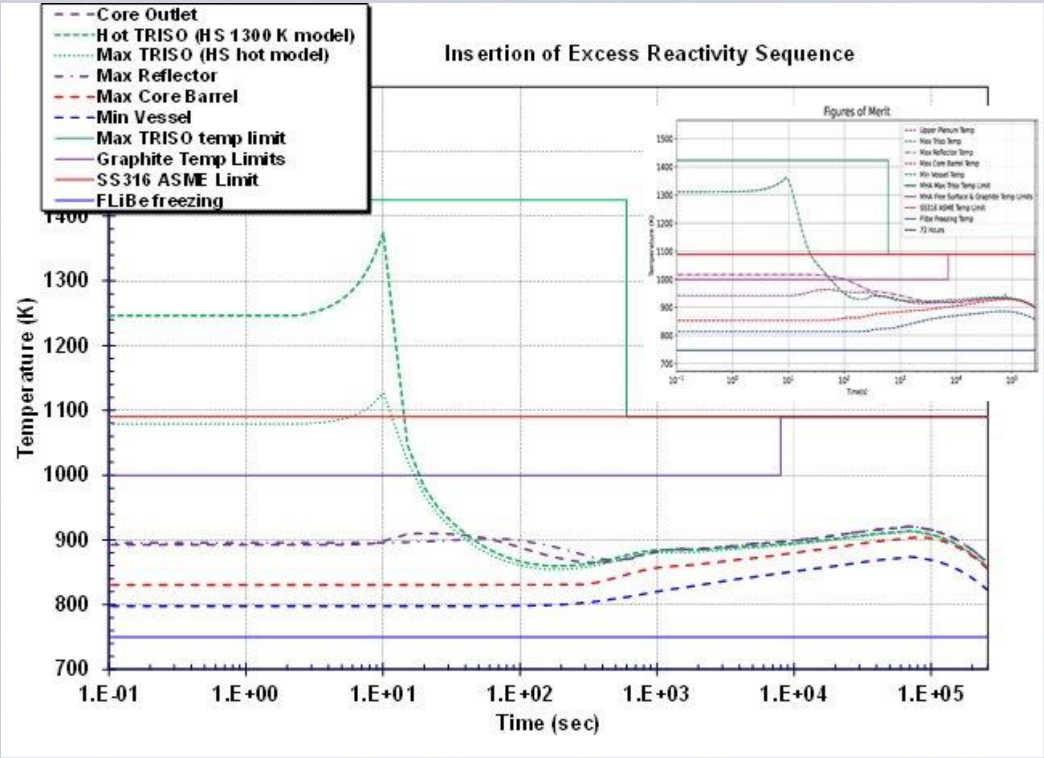
DHRS

PSAR Schematic

Severe Accident Analysis
Hermes I: SCALE/MELCOR Results

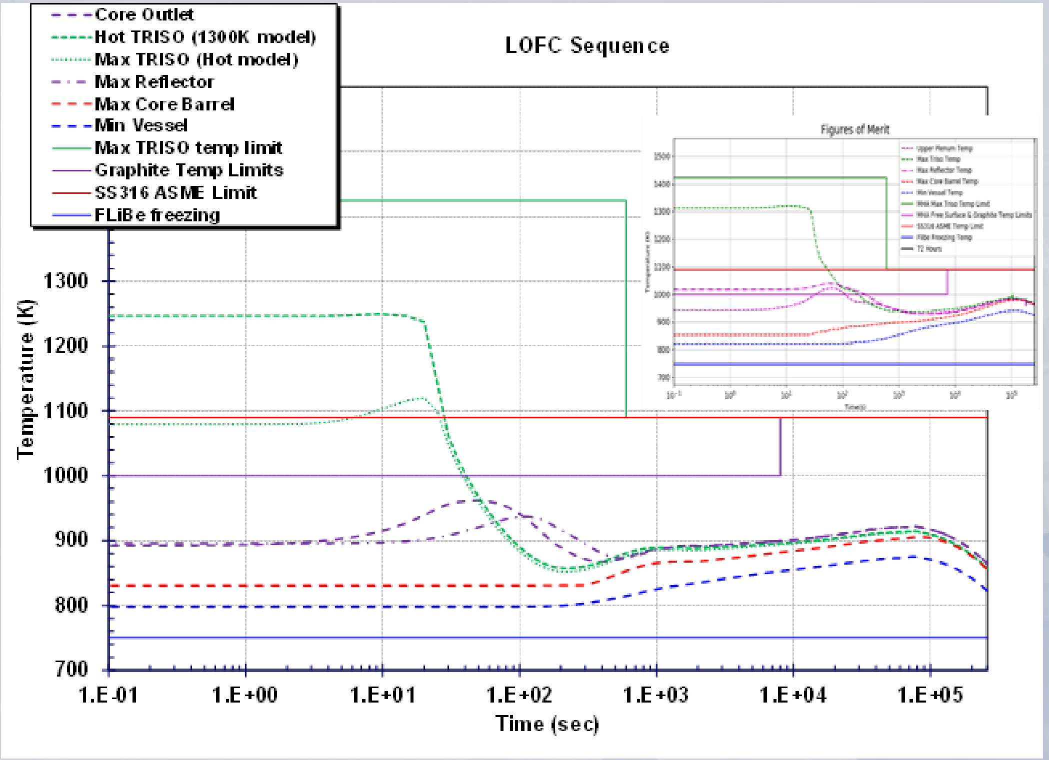
Insertion of Excess Reactivity

Withdrawal of control element inserts 3.02$ over 100 seconds
Reactor trips on high power



Loss of Forced Circulation

Concurrent trip of primary and intermediate coolant pumps
Reactor trips on overtemperature

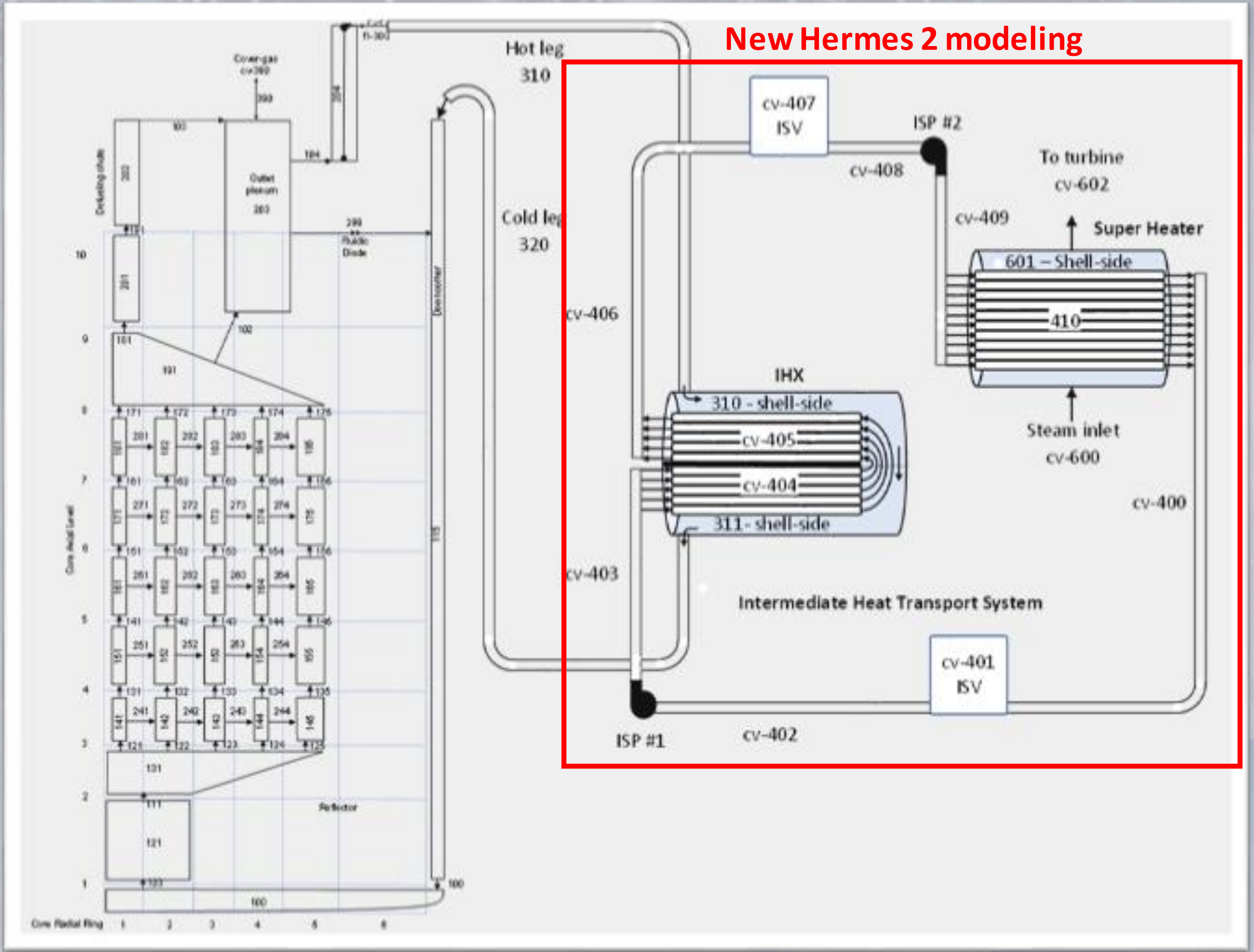


MELCOR results as compared with PSAR (upper right)

Severe Accident Analysis
Hermes 2 Construction Permit

- On September 11, 2023, the NRC staff accepted the Hermes 2 CP application for detailed review (ML23233A167)

- RES is supporting NRR's review
  - o Hermes 1 model updated to include the intermediate loop and the superheater steam heat exchanger
  - o Perform independent scoping analysis to understand differences in DBA response between Hermes 1 and Hermes 2



Severe Accident Analysis
Pebble-bed gas-cooled reactor – PBMR-400

Reactor Characteristics

- 400 MWth reactor, graphite moderated
- Helium-cooled & TRISO-particle pebble-fueled at 10 wt.% U-235
- Fuel discharged at high burnup (90 GWd/MTU)

New Modeling Capabilities

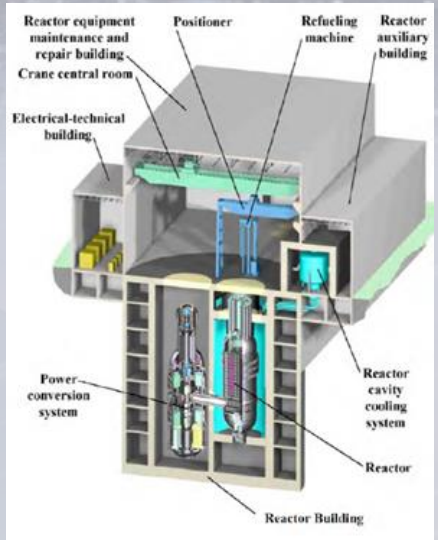
- SCALE: Interface for rapid depletion of TRISO fuel for efficient computational costs (*increased efficiencies for performing wide array of sensitivity studies)*
- MELCOR: TRISO fuel pebble thermal response, radionuclide diffusion, and failure models. Leveraged modeling efforts performed under NGNP (2006-2013)

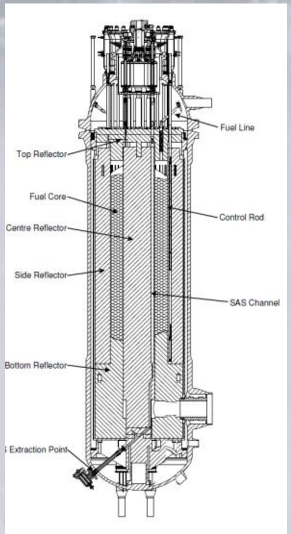
Accidents Modeled

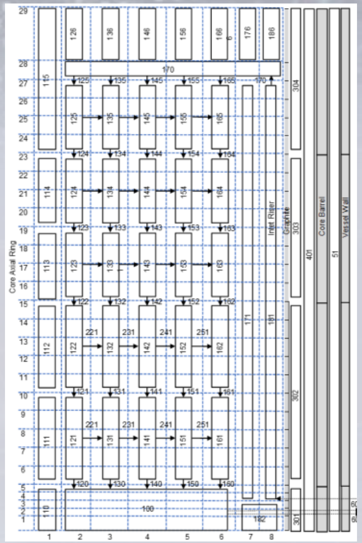
- Depressurized loss-of-forced circulation

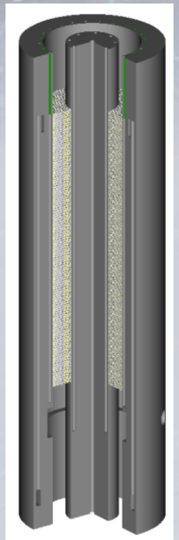
Insights

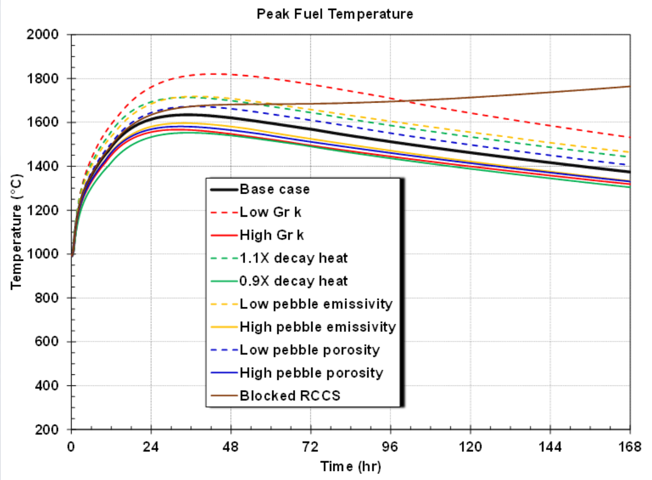
- Graphite oxidation from air ingress does not generate sufficient heat to impact fuel
- Passive heat dissipation into reactor cavity limits release from fuel failure
- A low graphite conductivity has the largest impact on the peak fuel temperature and release

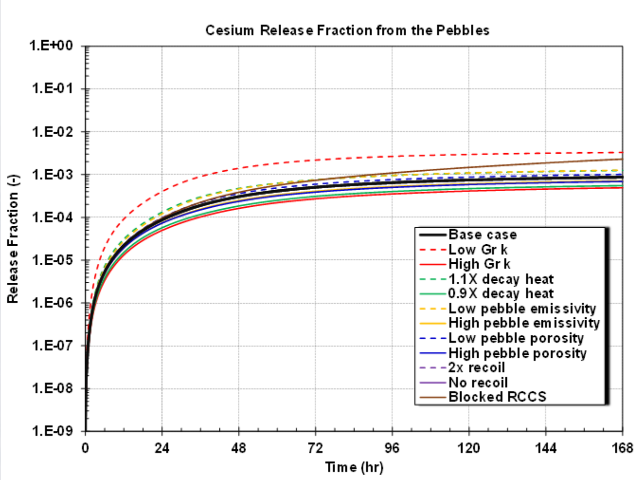












Severe Accident Analysis
Heat pipe reactor – INL Design A

RES

Reactor Characteristics

- 5 MWth with a 5-year operating lifetime
- 1,134 heat pipes fueled with metallic U (19.75 wt.% U-235)
- Reactivity controlled via control drums

New Modeling Capabilities

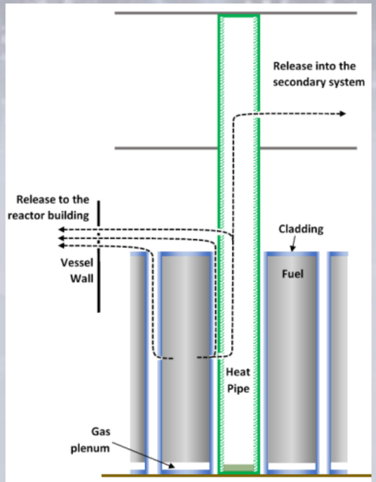
- SCALE: New 302-group fast-spectrum library & 3D visualization improvements (*rapid model generations)*
- MELCOR: New thermophysical properties of sodium and potassium added, new HP-specific model (includes HP working fluid, HP connection to the secondary heat exchanger, and various HP failure modes)

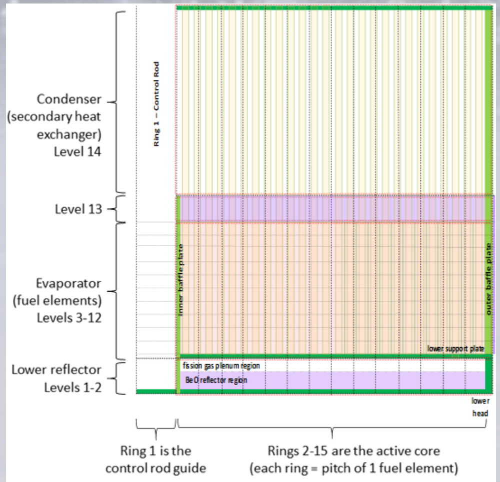
Accidents Modeled

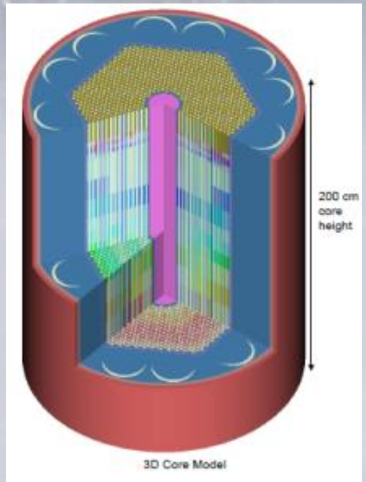
- Transient overpower (TOP), loss-of-heat sink, and anticipated transient w/o SCRAM

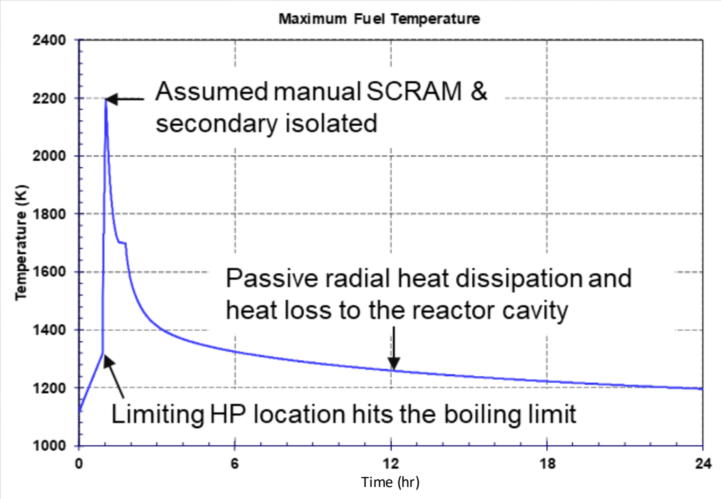
Key Insights

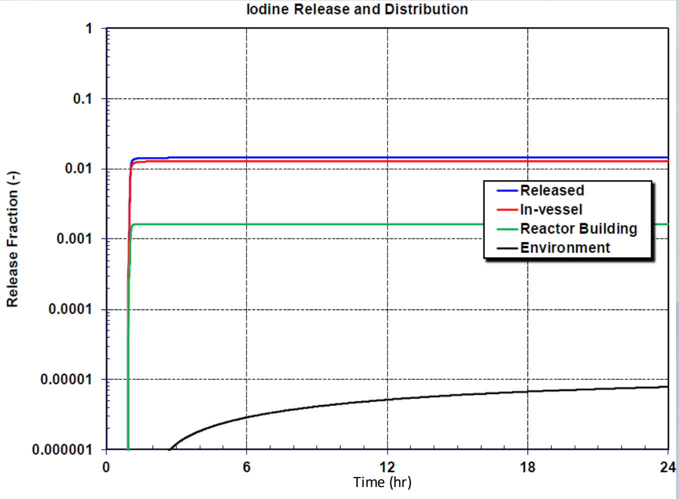
- Following SCRAM, passive heat dissipation into reactor cavity ends the release from fuel
- Heat pipe depressurization on failure drives the release from the reactor vessel into the reactor building
- Reactor building bypass requires two failures in a single heat pipe – one in the condenser region and another in the evaporator region











Severe Accident Analysis
Molten-salt-fueled reactor – MSRE

Reactor Characteristics

- 10 MWth reactor, graphite moderated at near atmospheric pressures
- Reactor fueled with dissolved fuel in molten salt (34.5 wt. % U-235)
- Fuel loop transit time ~25 seconds

New Modeling Capabilities

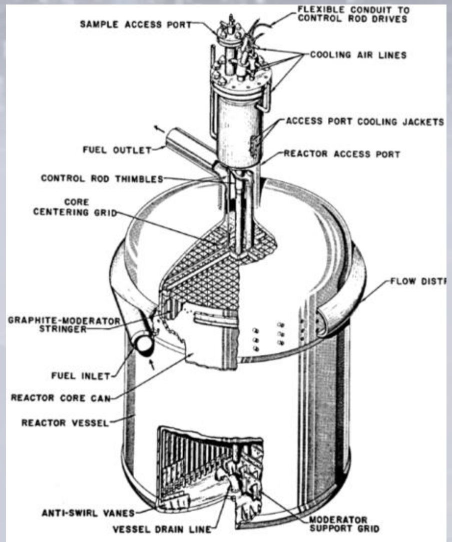
- SCALE: Modifications for handling liquid fuel, time-dependent system-average removal (e.g., simulating the off-gas system)
- MELCOR: Thermal hydraulic and equations of state for Flibe, Generalized Radionuclide Transport and Retention (GRTR) modeling framework, molten salt chemistry and physics pertaining to radionuclide transport, fluid fuel point kinetics

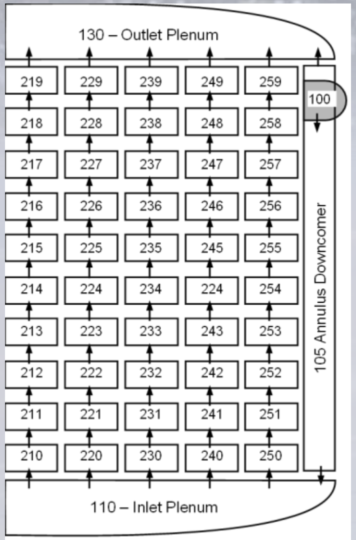
Accidents Modeled

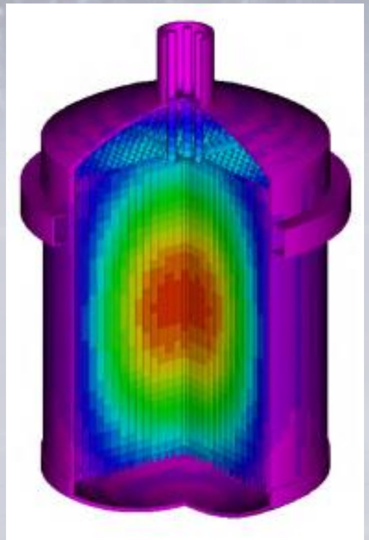
- Full reactor inventory molten salt spill (dry and wet conditions)

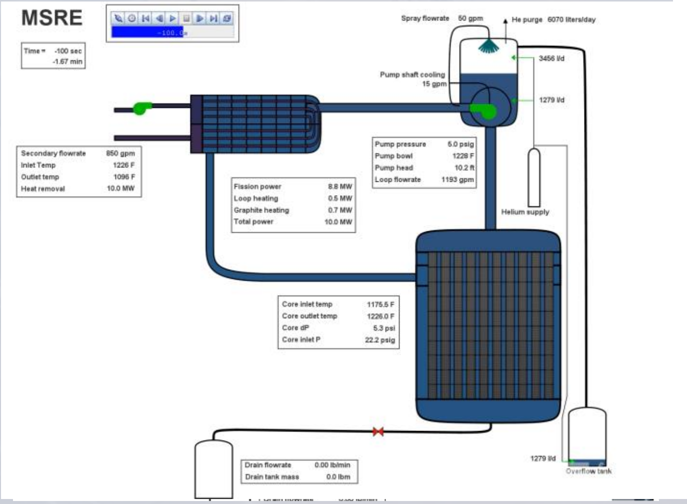
Key Insights

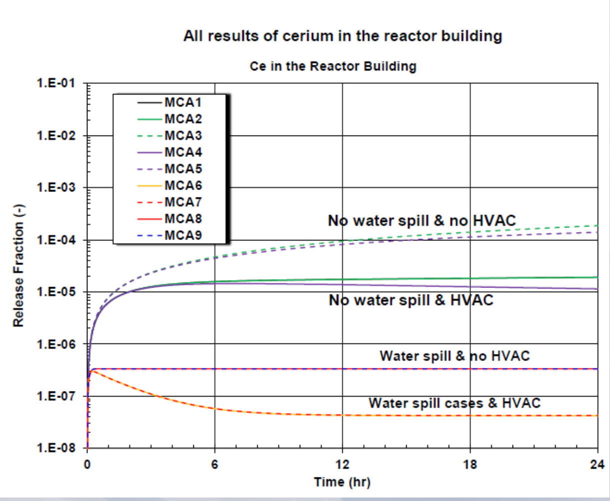
- Auxiliary filter operation increases the release of xenon to the environment while also filtering airborne aerosols
- Aerosol releases to the environment were small due to settling in the reactor cell, capture in the filter, and capture in the condensing tank in the water spill cases
- The aerosol mass in the reactor building also spanned many orders of magnitude depending on scenario assumptions











Severe Accident Analysis Sodium-cooled fast reactor – ABTR

Reactor Characteristics

- 250 MWth pool-type reactor, utilizing metallic U / HT-9 fuel rods
- Reactor fueled with U-Pu-Zr fuel slugs
- Liquid sodium coolant

New Modeling Capabilities

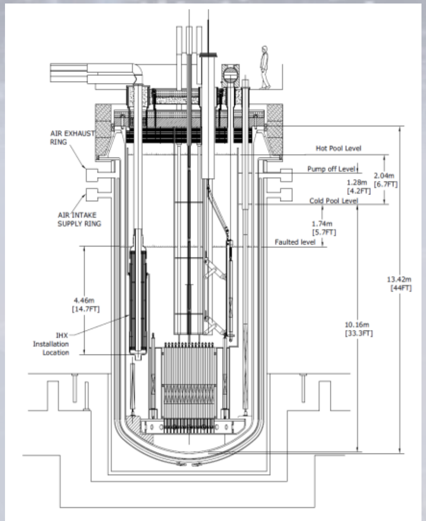
- SCALE: New capabilities in TRITON for generating nodal data for cartesian and hexagonal lattices and cells (e.g., few group homogenized cross-sections)
- MELCOR: SFR material properties, metallic fuel damage progression and radionuclide release models, sodium fire model

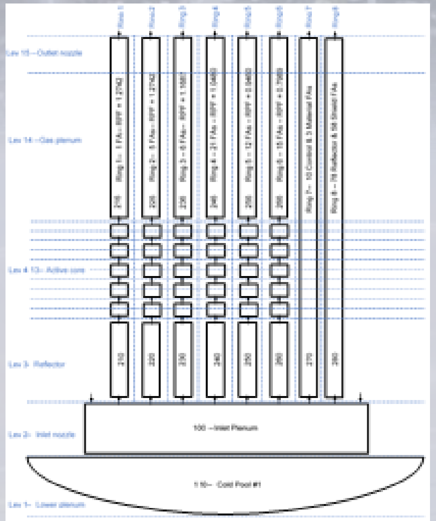
Accidents Modeled

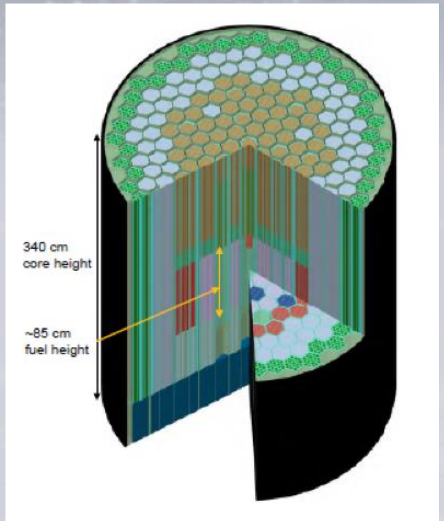
- Unprotected transient overpower, unprotected Loss-of-Flow (ULOF), and single blocked assembly

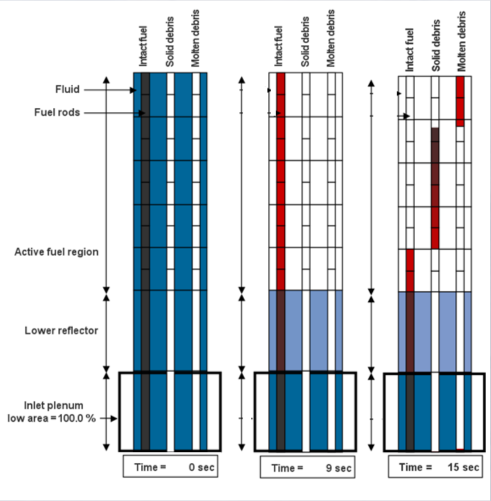
Key Insights

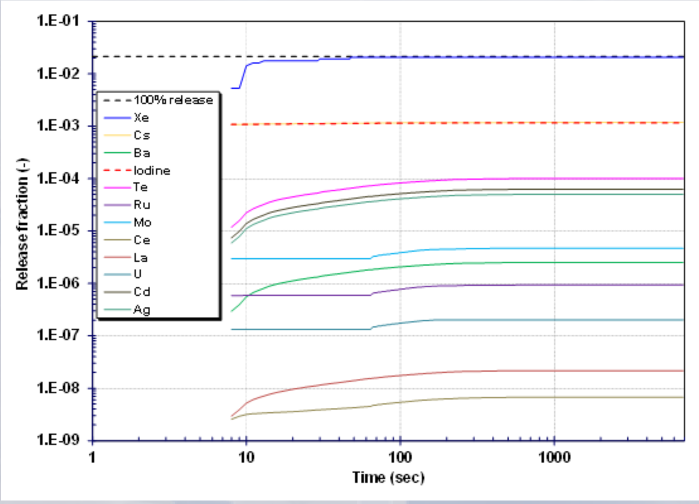
- With ULOF, core power eventually converges on the DRACS heat removal rate
- A single blocked assembly leads to rapid fuel melt











Severe Accident Analysis
MELCOR Validation & Verification Basis

RES

Leverage existing LWR Assessment Base

Code Documentation & Assessment

RN Physics / Transport
ABCOVE, ACE AA1, AA2, AA3, AHMED, CSE-A9, DEMONA, FALCON 1&2, LACE LA1 & LA3, LACE-LA4, Marviken ATT-4, Poseidon, RTF ISP-41, STORM, VANAM-M3, VERCORS, VI (ORNL)

Core Heatup & Degradation
CORA-13, DF-4, FPT1 & FPT3, LHF/OLHF, LOFT-FP2, MP1 & MP2 (SNL), PBF-SFD, Quench 11, VERCORS, VI (ORNL)

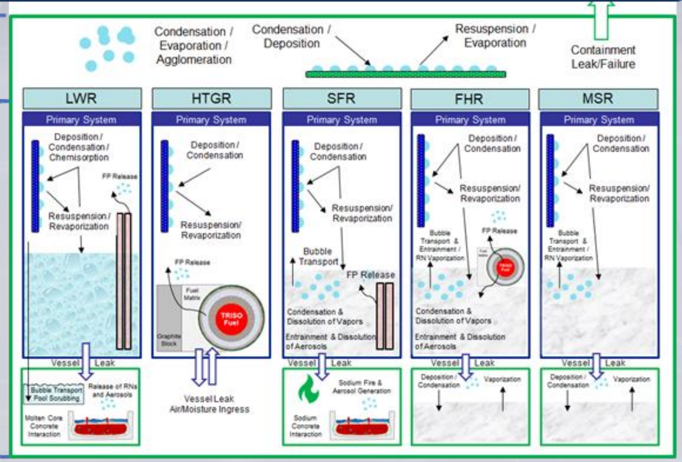
Integral / Accidents
FPT1 & FPT3, TMI-2, Fukushima

RN Release
VERCORS, VI (ORNL))

RPV & Primary TH
Bethsy, Flecht-Seaset, GE Level Swell, LOFT-FP, Marviken, Blowdown Tests, NEPTUN, RAS MEI

Ex-Vessel Debris
IET-DCH, OECD-MCCI, SURC

Containment
CSE-A9, CSTF Ice Condenser Test, CVTR, DEHBI, GE Mark IISuppression Pool, HDR E-11, HDR V44, IET 1 thogh IET7 and IET 9, JAERI Spray Tests, NST Hydrogen Burn, NUPEC M-7-1, M-8-1, M-8-2, PNL Ice Condenser Tests, Wisconsin flat plate



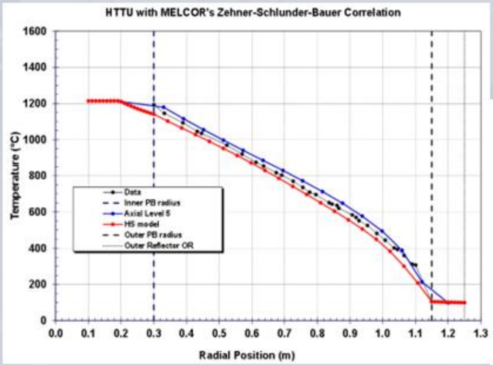
Non-LWR Specific Assessment Base

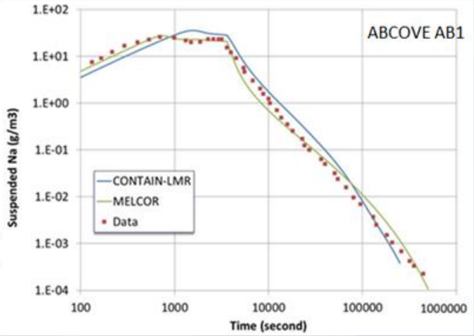
Test	Application
MSRE Zero-power coast-down	Molten Salt (Completed)
MSRE experiments	Molten Salt (Planned)
AB-1, AB-5, T-3	Sodium Fires (Completed)
LOF,LOHS,TOP, TREAT M-Series, ANL-ART-38	Sodium Reactors (Planned)
HTTU GEMINI	HTGR (Completed)
Air-Ingress Helical SG HT	HTGR (Planned)
PBMR-400 HTR-10	HTGR (In-progress)

TRISO Diffusion Release
IAEA CRP-6 Benchmark

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

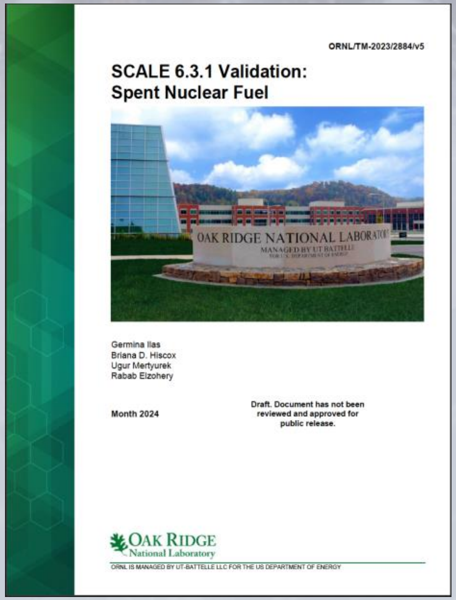
(1a): Bare kernel (1200 °C for 200 hours)
(1b): Bare kernel (1600 °C for 200 hours)
(2a): kernel+buffer+iPyC (1200 °C for 200 hours)
(2b): kernel+buffer+iPyC (1600 °C for 200 hours)
(3a): Intact (1600 °C for 200 hours)
(3b): Intact (1800 °C for 200 hours)





Severe Accident Analysis
SCALE Benchmarking & Validation Activities

SCALE Validation in Four Major Areas (Criticality Safety, Radiation Shielding, Reactor Physics, and Spent Fuel Inventory)



ORNL/TM-2023/2884/v5

SCALE 6.3.1 Validation: Spent Nuclear Fuel

Germina Ilas
Briana D. Hiscox
Ugur Mertyurek
Rabab Elzohery

Month 2024

Draft. Document has not been reviewed and approved for public release.

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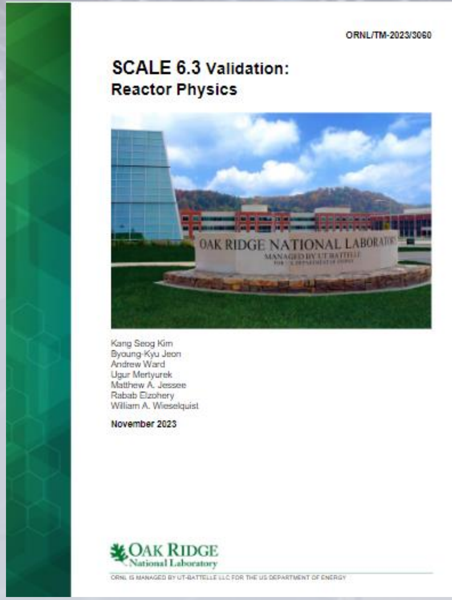

ORNL/TM-2023/3060

SCALE 6.3 Validation: Reactor Physics

Kang Seog Kim
Byoung-Kyu Jeon
Andrew Ward
Ugur Mertyurek
Matthew A. Jessee
Rabab Elzohery
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November 2023

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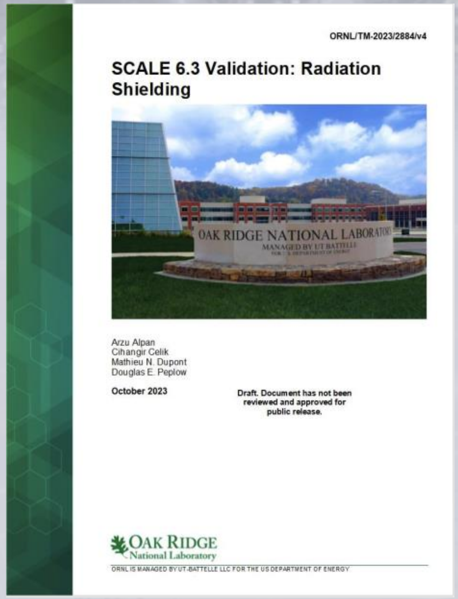

ORNL/TM-2023/2884/v4

SCALE 6.3 Validation: Radiation Shielding

Arzu Alpan
Cihangir Celik
Mathieu N. Dupont
Douglas E. Peplow

October 2023

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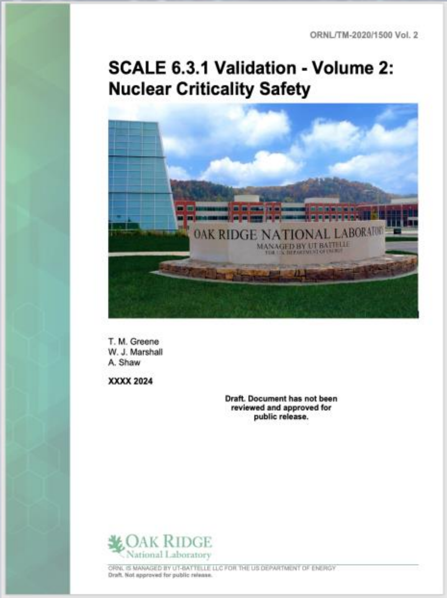

ORNL/TM-2020/1500 Vol. 2

SCALE 6.3.1 Validation - Volume 2: Nuclear Criticality Safety

T. M. Greene
W. J. Marshall
A. Shaw

XXXX 2024

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HTGRs

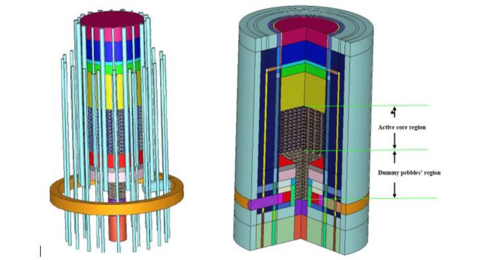

Figure 5.1. Illustration of HTR-10 benchmark model details.¶ (Channels in reflector regions [left], full reactor model [right]; images not to scale) [35].

Table 5.1. Eigenvalue results for high-fidelity HTR-10 benchmark.

	k_{eff}	σ	$\Delta k_{eff}{}^{a}$ (pcm)
Benchmark value [6]	1.00000	0.00370	reference
SCALE/KENO-VI CE ENDF/B-VII.1	1.00303 ± 0.00041	0.99661 ± 0.00031	303 ± 370
SCALE/KENO-VI CE ENDF/B-VIII.0	1.00604 ± 0.00027	0.99919 ± 0.00026	604 ± 370
SCALE/KENO-VI 252-group ENDF/B-VII.1	1.00265 ± 0.00031	0.99595 ± 0.00025	265 ± 370
SCALE/KENO-VI 252-group ENDF/B-VIII.0	1.00376 ± 0.00027	0.99746 ± 0.00025	376 ± 370

[a] Calculated as 10^5(k-eff$_{calculated}$ – k-eff$_{benchmark}$).

MSRs

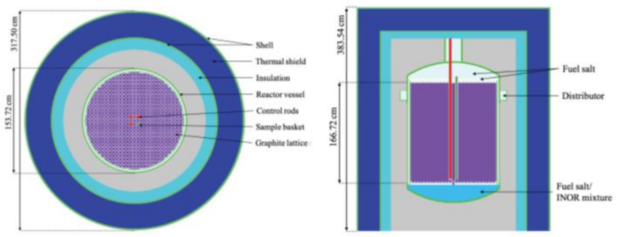

Figure 5.5. Cross-sectional illustrations of MSRE benchmark models [60].

Table 5.3. Eigenvalue results for the high-fidelity MSRE benchmark.

	k_{eff}	σ	$\Delta k_{eff}{}^{a}$ (pcm)
Benchmark value	0.99978	± 0.00420	reference
SCALE 6.3.1/Shift CE ENDF/B-VII.1	1.019016	± 0.00010	1924 (± 420)
SCALE 6.3.1/Shift CE ENDF/B-VIII.0	1.021833	± 0.00010	2205 (± 420)

[a] Calculated as 10^5(k-eff$_{calculated}$ – k-eff$_{benchmark}$).

SFRs

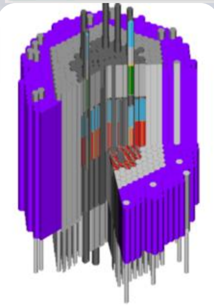

Figure 5.7. EBR-II SCALE model [62].

Table 5.4. Eigenvalue results for the high-fidelity EBR-II benchmark.

	k_{eff}	σ	$\Delta k_{eff}{}^{a}$ (pcm)
Benchmark value [7]	1.00927	± 0.00618	reference
SCALE 6.3.1/KENO-VI CE ENDF/B-VII.1	1.00722	± 0.00010	-205 (± 618)
SCALE 6.3.1/KENO-VI CE ENDF/B-VIII.0	1.00691	± 0.00013	-236 (± 618)

[a] Calculated as 10^5(k-eff$_{calculated}$ – k-eff$_{benchmark}$).

Severe Accident Analysis Summary and Next Steps

RES

1. Modeling gaps addressed through source code changes, phenomenological model development, and new analysis workflows in SCALE and MELCOR

2. SCALE & MELCOR models leveraged for supporting NRR's review of the Hermes Construction Permit Applications

3. Additional Code Enhancements & Capabilities In-Progress
   - Integration of SCALE/ORIGEN module into MELCOR for higher fidelity MSR transient analyses
   - Capability to model multiple working fluids
   - Functionality for horizontal heat pipe reactors
   - Refinement of specialized models (e.g., fluid freezing and cascading heat pipe failures)
   - Fission product chemistry refinement
   - Spatial dependence of reactivity feedback in SFRs

4. Data Needs
   - Validation – Criticality and depletion benchmarks that are representative of fuel designs and conditions, diffusivity of fission products, heat and mass transfer in diverse working fluids, etc.

SCALE & MELCOR code improvements and demonstration workshops have shown NRC is ready to support licensing reviews.

Consequence Analysis

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Division of Systems Analysis

Office of Nuclear Regulatory Research

MACCS Code Development Approach Summary

- Staff expects to complete most non-LWR Severe Accident Consequence Analysis Computer Code Development Plan tasks by Quarter 4 of FY24.

- Staff determined the resolution of the code development plan by identifying and adopting state-of-practice methods commonly used in the relevant topical area.

- Staff has concluded that enhancing the MACCS code on a generic basis for several tasks is not practical due to the requirement for detailed information regarding the chemical composition of the atmospheric source term.

MACCS Code Development Activities Status

RES

Phenomenological Areas	Fiscal Year						Reports	Notes
	2019	2020	2021	2022	2023	2024		
Nearfield Modeling	X	X	X				SAND2020-2609 SAND2021-6924	MACCS 4.1 has implemented upgraded nearfield models
Radionuclide Release Screening			X	X			SAND2021-11703 SAND2022-12018	MACCS 4.2 has increased the radionuclide limit to 999
Radionuclide Size, Shape, and Chemical Form				X			SAND2022-12766	MACCS deposition and dosimetry capabilities are state-of-practice
Tritium Modeling				X	X	X	SAND2022-12016	MACCS can offer conservative estimates for tritium inhalation pathways. Staff will update the tritium inhalation dose coefficient in the MACCS code to include skin absorption. Tritium ingestion pathways may be addressed using alternative codes.
Radionuclide Evolution in Atmosphere					X	X	In progress	State-of-practice models for generic reactive atmospheric transport are limited in availability
Decontamination Modeling							Not started	MACCS decontamination modeling shows no specific nexus to non-LWR technologies
Chemical Hazards							Not started	Chemical hazards may be out of scope for severe accident probabilistic consequence analysis

MACCS Code Development Activities Path Forward (1/2)

- Staff will continue coordinating with MELCOR code developers to determine whether new source terms necessitate MACCS model enhancements
  - Designs with significant gaseous releases may benefit from a state-of-practice resistance model for deposition
  - Designs that have the potential for large releases of tritium as HT gas or for releases leading to significant ingestion doses may require updates to MACCS
- Staff may not pursue two tasks as part of the non-LWR Severe Accident Consequence Analysis Computer Code Development Plan
  - Decontamination modeling shows no specific connection to non-LWR technologies
  - Chemical hazards may be out of scope for severe accident probabilistic consequence analysis

MACCS Code Development Activities Path Forward (2/2)

- Several tasks identified potential follow-on work that may benefit both non-LWR and LWR technologies
- Tasks will be pursued in active code maintenance, documentation, and state-of-practice development activities. Examples include:
  - Benchmarking and stress-testing MACCS wake effect and downwash models
  - Incorporate EPA PRIME model plume rise/downwash algorithms
  - Examine sensitivity of FGR13 dose coefficients to alternate chemical forms
  - Benchmark MACCS regression-based deposition model against AERMOD/HYSPLIT resistance-based model
  - Upgrade MACCS dose coefficient file to allow user specified FGR13 chemical forms
  - Upgrade MACCS deposition model to incorporate state-of-practice resistance model for deposition
  - Update guidance for modeling consequences of tritium releases when using MACCS

MACCS Non-LWR Code Demonstration Project

RES

- Purpose:
  - Provide practical test of the capabilities of the MACCS code to analyze a selected conceptual advanced reactor design under a postulated accident scenario (ADAMS Accession No. ML23045A044)
- Conclusions:
  - Staff confirmed that, despite some limitations, analysts can use the flexibility of the MACCS code to analyze the offsite consequences of an advanced reactor design under a postulated accident scenario
  - The evaluation exercise provided valuable practical experience in implementing new ORIGEN inventories and MELCOR source terms in MACCS
- Candidates for future research activities:
  - Examine methods to analyze or conservatively bound accidents with simultaneous release and fission.
  - Continue evaluating radionuclide importance to dose for non-LWR inventories and expand these evaluations to include ingestion doses
  - Use core radionuclide inventory and atmospheric release from example SCALE and MELCOR demonstration calculations for further MACCS code demonstrations to facilitate NLWR knowledge management for NLWR consequence assessments

Conclusions

RES

- MACCS was originally designed with flexibilities to accommodate various types of facilities.

- Staff considers MACCS code readiness adequate for assessing consequences associated with non-LWR technologies.

- MACCS code demonstration projects present opportunities to enhance knowledge management for conducting consequence assessments, both for non-LWR and LWR applications.

Near-Field Transport

RES

Improve MACCS near-field atmospheric transport and dispersion capability to better treat building wake effects in the near field (<500 meters from a containment or reactor building) given the need for probabilistic dose calculations closer to non-LWRs relative to large LWRs.

- Status: Complete
  - The assessment concluded that MACCS 4.0 can be used conservatively at distances significantly shorter than 500 meters downwind from a containment or reactor building.
  - MACCS v4.1 includes additional capabilities to better account for the nearfield wake and meander effects using the Ramsdell and Fosmire wake/meander model or the Regulatory Guide 1.145 wake/meander model.
- Next Steps: None
- Source Term Monitoring and Coordination: No
- Potential Future Work:
  - Consider benchmarking and stress-testing MACCS wake effect and downwash models. This task may be considered part of standard MACCS code validation and verification activities.
  - Consider incorporating the U.S. Environmental Protection Agency PRIME model plume rise/downwash algorithms. This task may be considered part of normal MACCS code development activities.

Radionuclide Screening

RES

Perform a screening analysis to identify which subset of radionuclides to include in MACCS calculations for each non-LWR type given the different mix of radionuclides that may be released in accidents from each type.

- Status: Complete
  - Staff developed a quantitative method for identifying radionuclides of potential interest for advanced reactors. The method, which is consistent with the approaches used to identify radionuclides for consideration for LWR consequence analyses, accounts for half-life, biological hazard, and relative abundance of radionuclides in the core.
  - In MACCS v4.2, the number of radionuclides that can be modeled was increased from 150 to 999. This enhancement enables the modeling of all 825 nuclides for which dose coefficients are available from Federal Guidance Report (FGR)-13.
- Next Steps: None
- Source Term Monitoring and Coordination: No, releases of radioactivity in chemical forms different from those assumed in the MACCS DCF file (typically 1 µm AMAD oxides and hydroxides) may require the application of a suitable dose coefficient inhalation clearance class for the expected chemical/physical form in the environment.
- Potential Future Work:
  - Consider providing guidance to model all nuclides for which dose coefficients are available. This task may be considered as part of standard MACCS code documentation activities to update NUREG/CR-7270 (ML22294A091).
  - Recommend coordinating inventory file processing with MELCOR inventory file processing.
  - Consider quantitative screening of additional advanced reactor inventories and ingestion pathway radionuclide screening.

Radionuclide Size, Shape, and Chemical Form

Evaluate potential differences in radionuclide releases from non-LWRs relative to LWRs including different aerosol size distributions, shape factors, and chemical forms. Based on the evaluation, improve MACCS capabilities for atmospheric transport and dosimetry to appropriately capture these issues for probabilistic consequence analysis. If necessary, consider a state-of-practice resistance model for dry deposition.

- Status: Complete
  - Current MACCS capabilities for deposition modeling appear to be consistent with the state of practice for particulate wet and dry deposition.
  - The dosimetry model in MACCS aligns with the state of practice. MACCS's code capabilities for dosimetry can accommodate variable chemical forms by employing alternative dose coefficients derived from FGR-13.
- Next Steps: None.
- Source Term Monitoring and Coordination: Yes, releases of radioactivity in chemical forms other than those assumed in the MACCS DCF file (typically 1 µm AMAD oxides and hydroxides) may require modification of the MACCS DCF file by either the MACCS code developer or by the MACCS code user.
- Potential Future Work:
  - Consider improving documentation of physical and chemical forms assumed for developing DC file.
  - Consider examining sensitivity of FGR13 DCs to alternate chemical forms.
  - Consider modifying MACCS/MACCS DC file to allow user specified FGR13 chemical forms.
  - Consider benchmarking MACCS regression-based deposition model against AERMOD/HYSPLIT resistance-based model.
  - Consider upgrading MACCS deposition model to incorporate state-of-practice resistance model.

Tritium Modeling

RES

Develop MACCS model and/or dosimetry updates to better account for the unique behavior of tritium which is very mobile and can enter biological systems as part of water and organic molecules.

- <u>Status</u>: Complete
  - MACCS is capable of modeling inhalation doses resulting from tritium released as water vapor (HTO), but it may overestimate inhalation doses (compared to UFOTRI and ETMOD) from tritium released as hydrogen gas (HT) by approximately two orders of magnitude. Doses from inhalation of HT or HTO releases may remain low unless large amounts of tritium are released.
  - MACCS is not currently suited to modeling ingestion doses arising from tritium releases, but doses from ingestion of tritium incorporated into foodstuffs may also be low unless large quantities of tritium are released.
- <u>Next Steps</u>: Staff recommends updating the tritium inhalation dose coefficient in the MACCS DCF file to include the standard 50% supplement for uptake via skin absorption during air immersion.
- <u>Source Term Monitoring and Coordination</u>: Yes. Designs with the potential for large tritium releases as HT gas or releases leading to significant ingestion doses may require either an update to MACCS or a tritium-specific consequence code such as UFOTRI or ETMOD.
- <u>Potential Future Work</u>:
  - Consider updating guidance for modeling consequences of tritium releases when using MACCS and ingestion doses from large releases using codes such as UFOTRI or ETMOD. This task may be considered as part of the standard MACCS code documentation activities to update NUREG/CR-7270 (ML22294A091).
  - Staff will rely on the results of source term monitoring and coordination and input from program office staff to determine whether the resources needed to upgrade the MACCS food model are justified in the future. It may be noted that integration of a tritium-specific food model may be a major effort.

Radionuclide Evolution in the Atmosphere

RES

Identify whether non-LWR accident releases may be more subject to evolution in the atmosphere relative to LWR releases based on differences in hygroscopic properties or potential for chemical reactions during transport

- Status: In progress
  - Staff completed a literature review to comprehend the potential chemical and physical transformations and their modeling approaches in other state-of-the-art codes for atmospheric transport, diffusion, and deposition. Notable codes encompassing these transformations are HYSPLIT, CMAQ, WRF-CHEM, SORAMI, and RATCHET .
  - Staff is evaluating the feasibility and methodology for MACCS to simulate these potential atmospheric transformations. Additionally, staff is planning a model intercomparison exercise against codes that simulate the transformation of iodine to assess the dosimetry significance of chemical and physical atmospheric evolution.
- Next Steps: Staff expects that transformation kinetics may vary significantly for individual chemical forms, such as UF6, to the extent that generic code updates may not adequately address highly reactive species.
- Source Term Monitoring and Coordination: Yes, releases of radioactivity in chemically reactive forms may require chemical-form specific transport and dispersion modeling.
- Potential Future Work: Source term monitoring and coordination efforts will continue to identify design-specific chemical and physical forms requiring code updates via the normal MACCS code development cycle.

Decontamination Modeling

RES

Based on the potential for non-LWRs to be sited closer to developed/urban lands, develop updated decontamination costs, durations, and dose reduction factors to account for the differences in decontaminating more urban areas relative to the generally rural areas where most large LWRs are sited.

- <u>Status</u>: Not started.
- <u>Next Steps</u>: No additional work is scheduled for non-LWR code development in this area due to the specific nexus to non-LWR technologies and the availability of a method to address variations in decontamination between urban and rural areas.
- <u>Source Term Monitoring and Coordination</u>: No.
- <u>Potential Future Work</u>: Conduct sensitivity analyses using existing MACCS decontamination cost model to examine sensitivity to differences in land use (e.g., population density). This task may be considered as part of the standard MACCS code documentation and development activities.

Chemical Hazards

RES

Identify whether non-LWRs themselves, or because of their potential collocation with industrial processing plants, create greater likelihood of chemical releases to the environment. If appropriate, update MACCS to integrate CHEM_MACCS for probabilistic calculations of offsite consequences of chemical releases.

- <u>Status</u>: Not started.
- <u>Next Steps</u>: No additional work is scheduled for non-LWR code development in this area due to the specific nexus to non-LWR technologies. Furthermore, any chemical hazard would be design- and source term specific.
- <u>Source Term Monitoring and Coordination</u>: Yes, if chemical hazards are found to be within scope for severe accident consequence analysis.
- <u>Potential Future Work</u>: None. However, staff could leverage methods and lessons learned from the development of CHEM_MACCS to identify necessary MACCS model updates for probabilistic calculations of offsite consequences of chemical releases. This task may be considered as part of the standard MACCS code development activities.

Update on Volume 4 – Licensing and Siting Dose Assessment Codes

John Tomon

Chief, Radiation Protection Branch

Division of Systems Analysis

Office of Nuclear Regulatory Research

Volume 4:
Licensing and Siting Dose Assessment Codes

- Tasks
  1. Consolidate/Modernize Dose Assessment Codes.
  2. Improve characterization of Source Terms.
  3. Improve Atmospheric Transport & Dispersion (ATD) Models.
  4. Update Dose Coefficient values.
  5. Develop Environmental Pathway Models.


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Protecting People and the Environment

Revision 1
March 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 4 – *Licensing and Siting Dose Assessment Codes*

Licensing and Siting Dose Assessment Codes

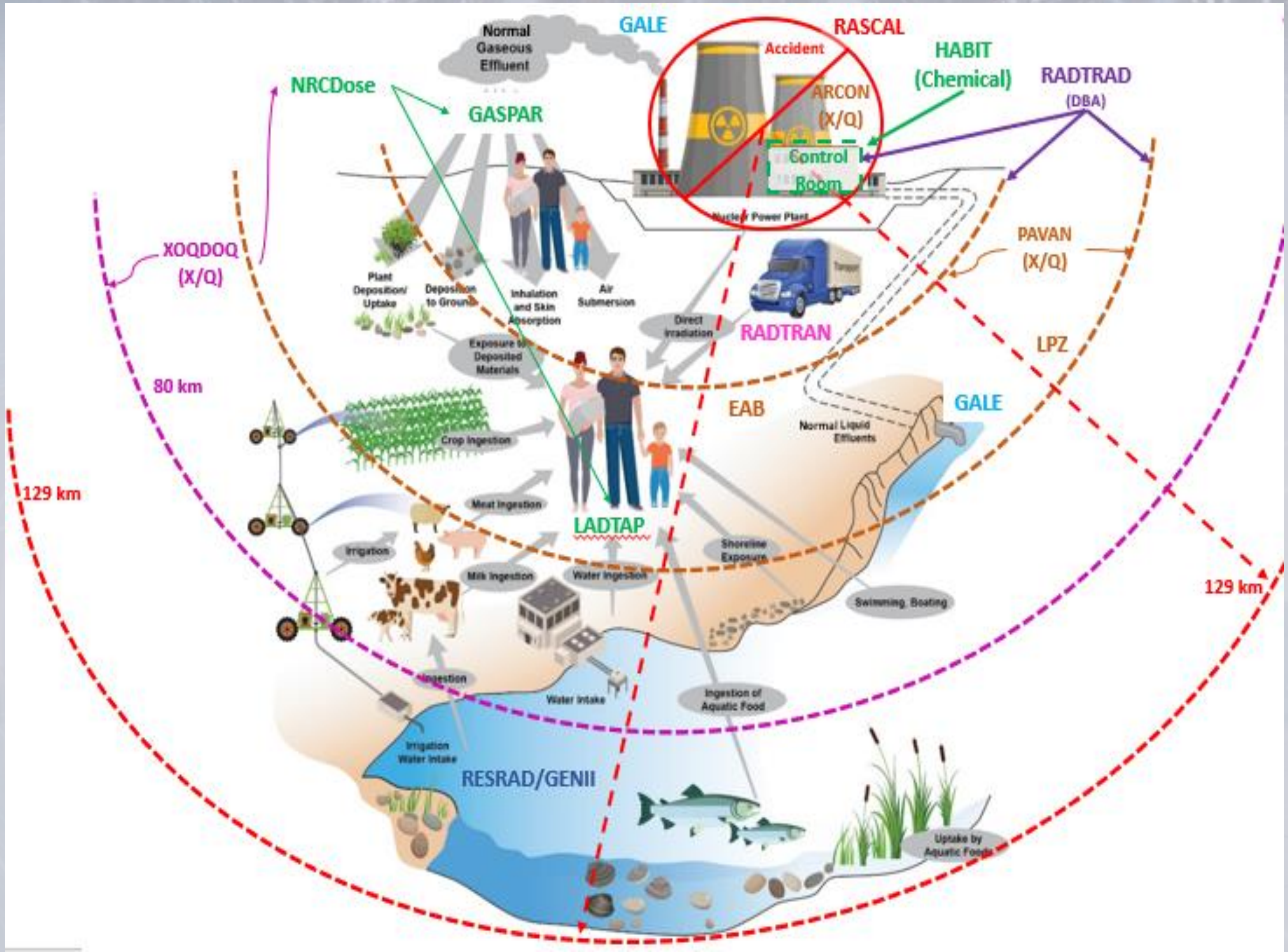
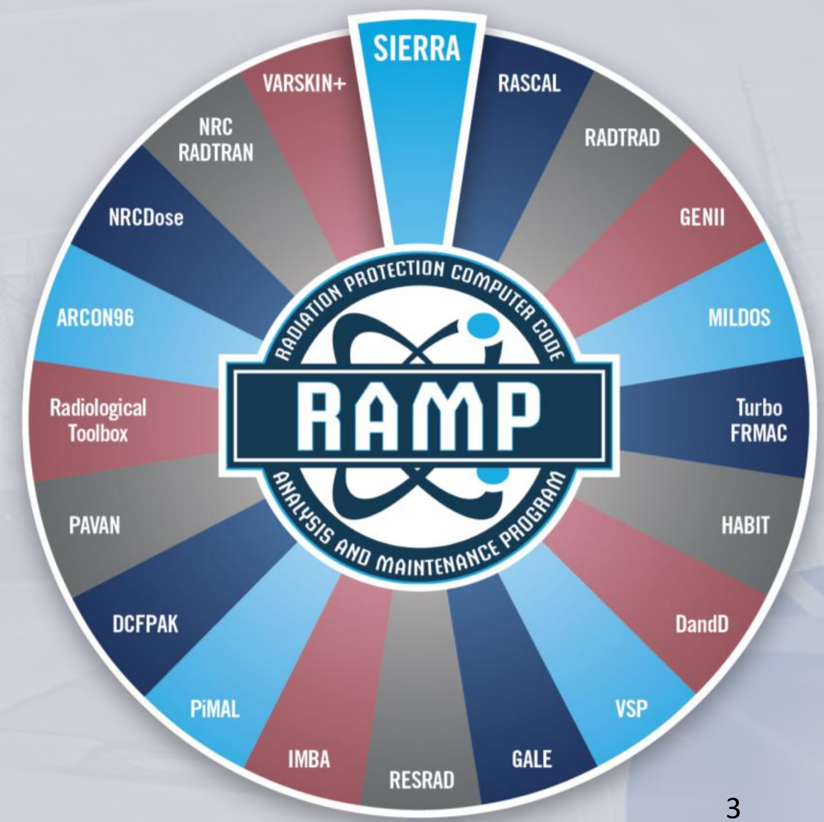


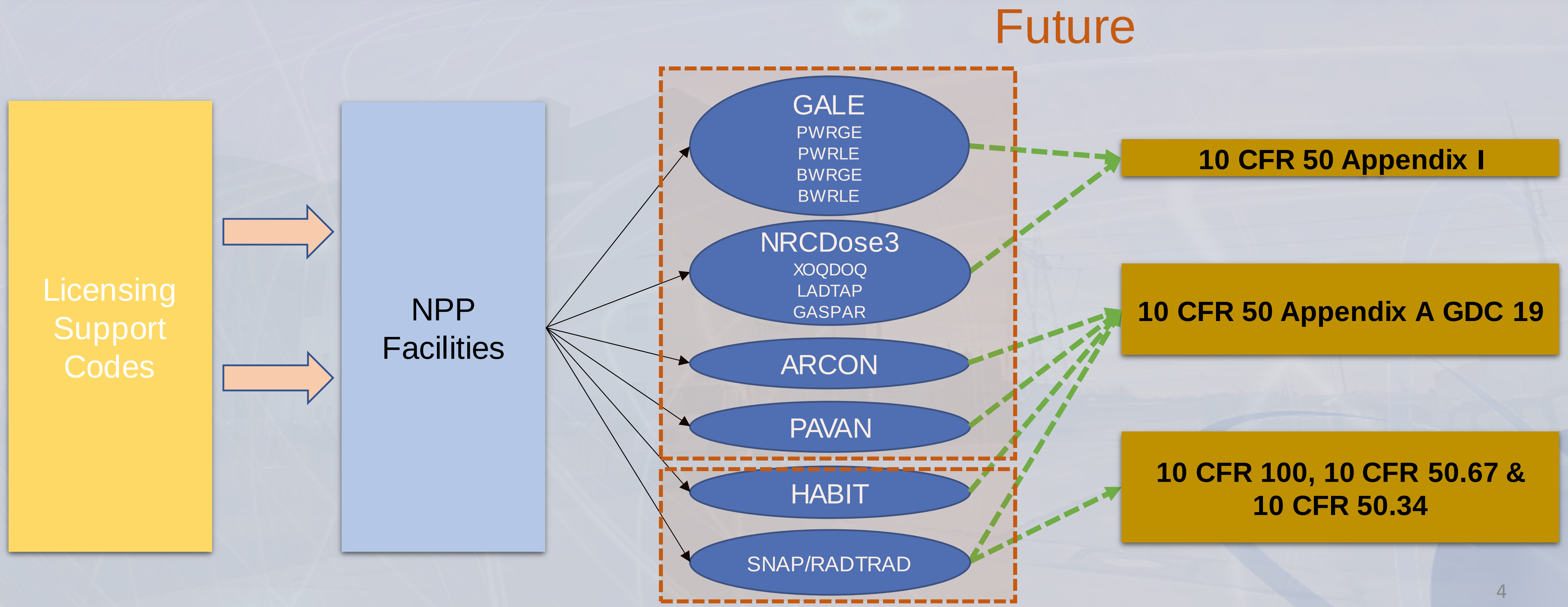
Image adapted from BNWL-1754, Models and Computer Codes for Evaluating Environmental Radiation Doses.

Over 10 codes used for NPP licensing and siting based on various regulations.



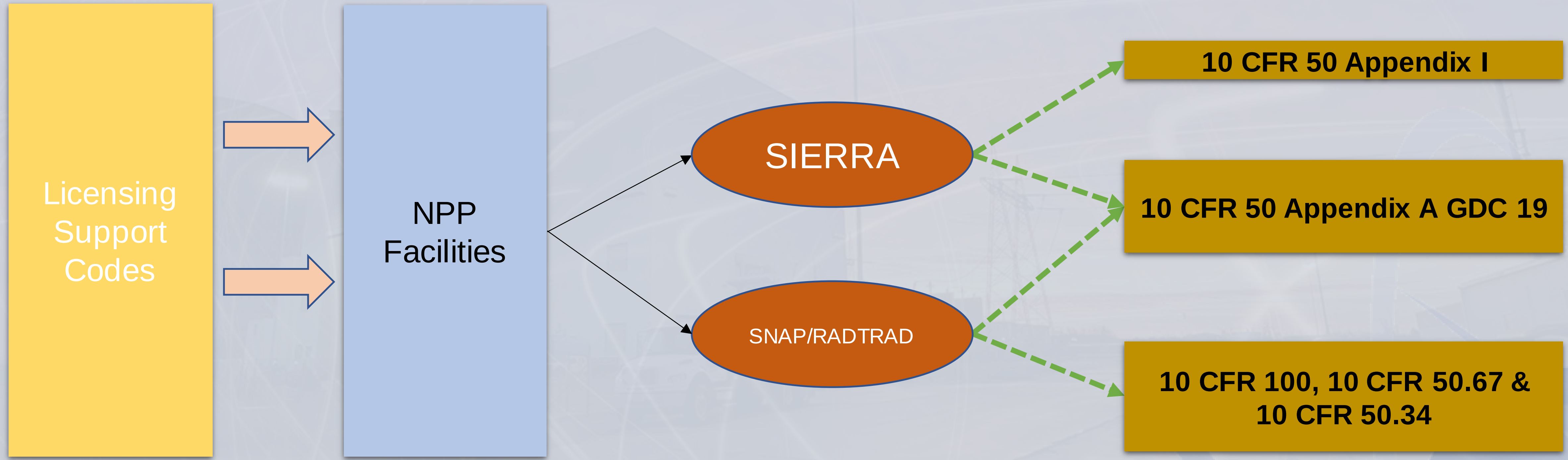
Licensing and Siting Code Regulations (1/2)

Current (Prior to Code Consolidation)



Licensing and Siting Code Regulations (2/2)

Future (Code Consolidation)



Accomplishments

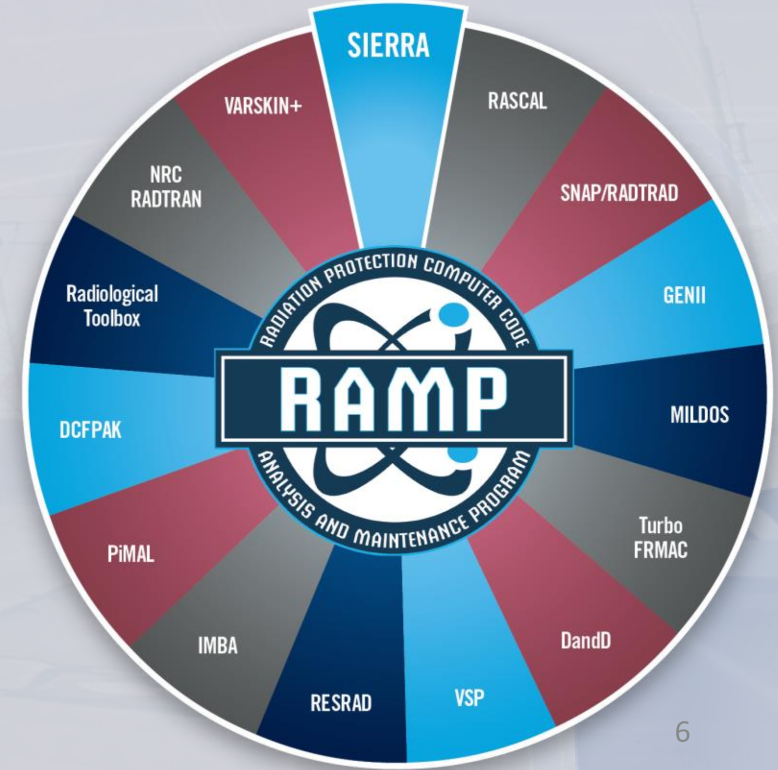
RES

- Task 1: Code Consolidation and Modernization.
- Task 2: Improve characterization of Source Terms (Phase 1).
- Task 3: Improve Atmospheric Transport & Dispersion (ATD) Models.

SIERRA

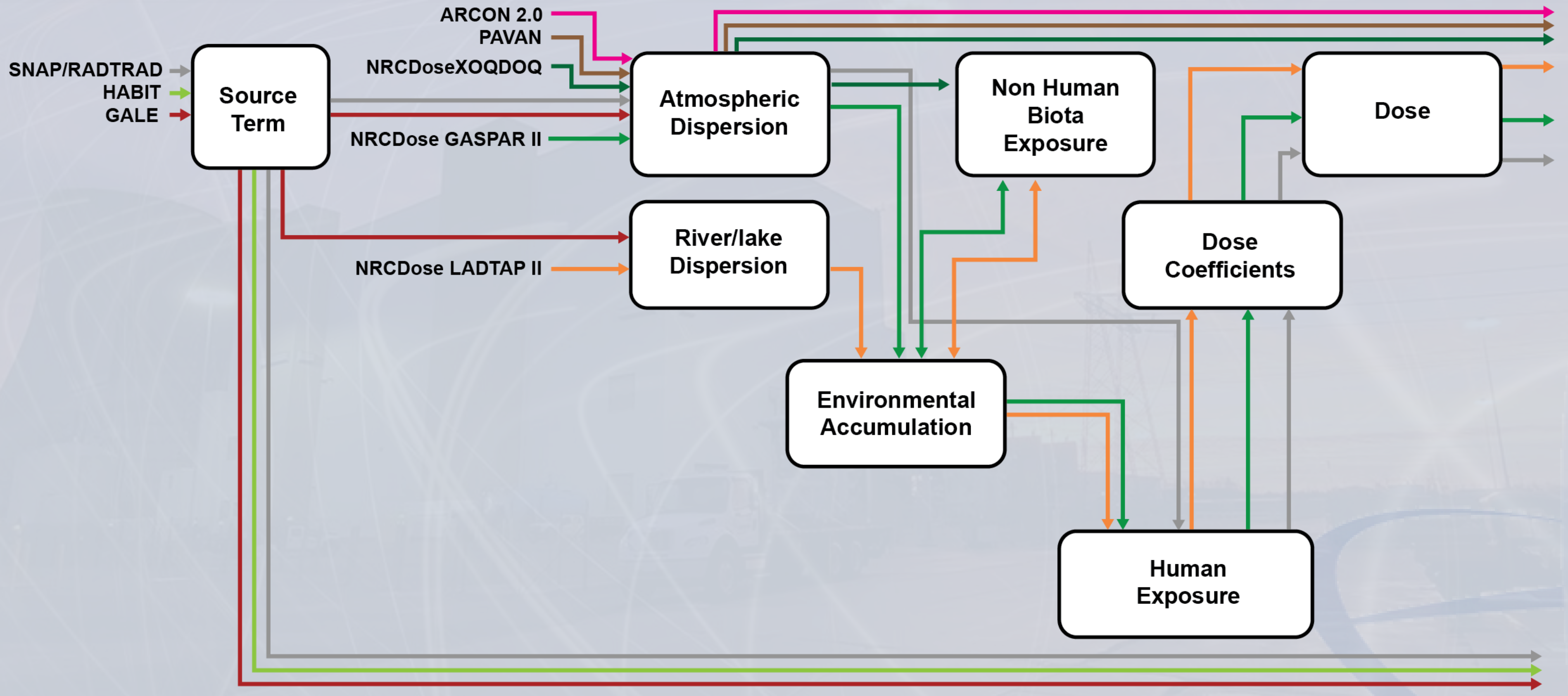
RAMP

Software Integration for Environmental Radiological Release Assessments



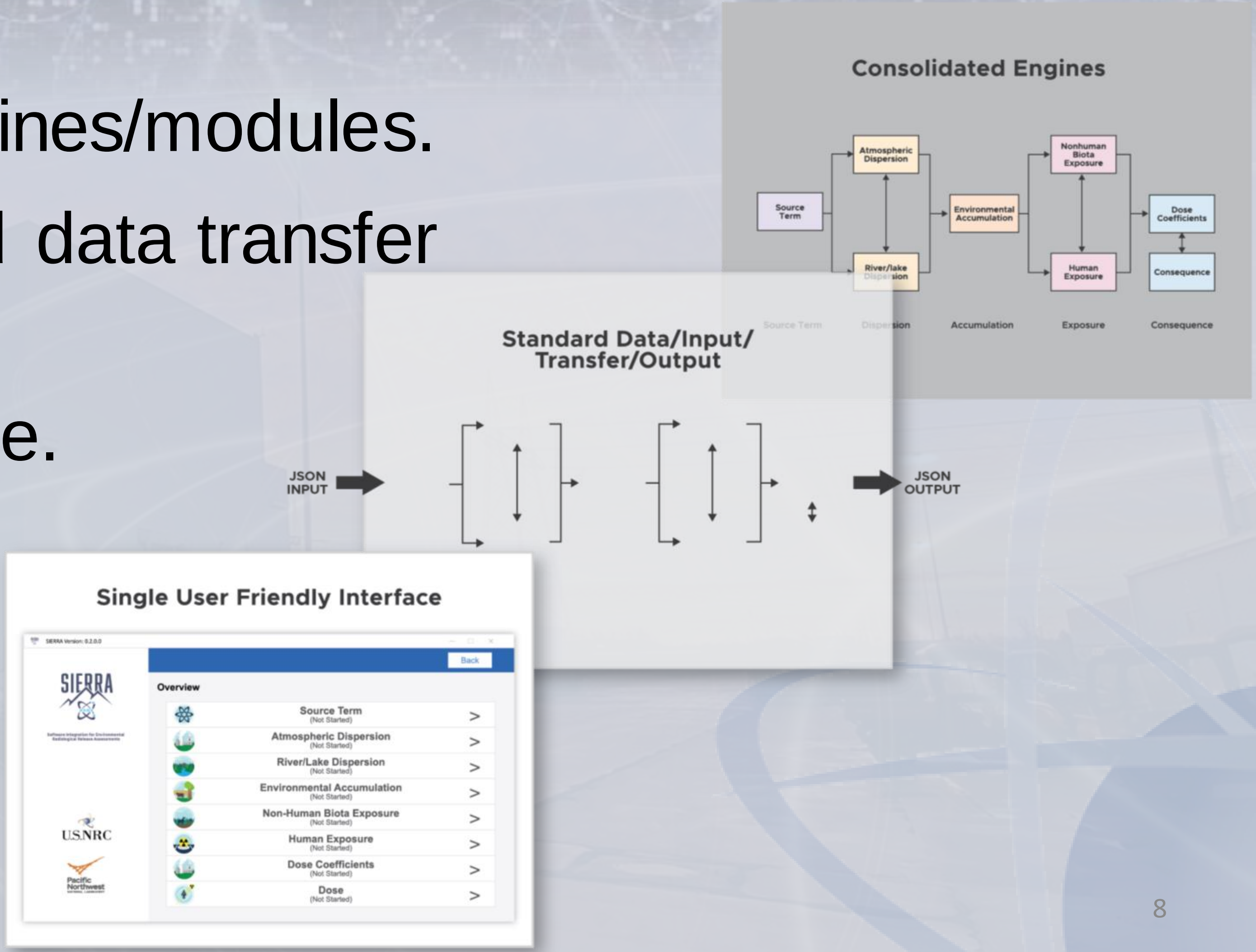
Task 1: Code Consolidation and Modernization (1/3)

Conceptual Model for the Consolidated Code 8 Modules/Engines



Task 1: Code Consolidation and Modernization (2/3)

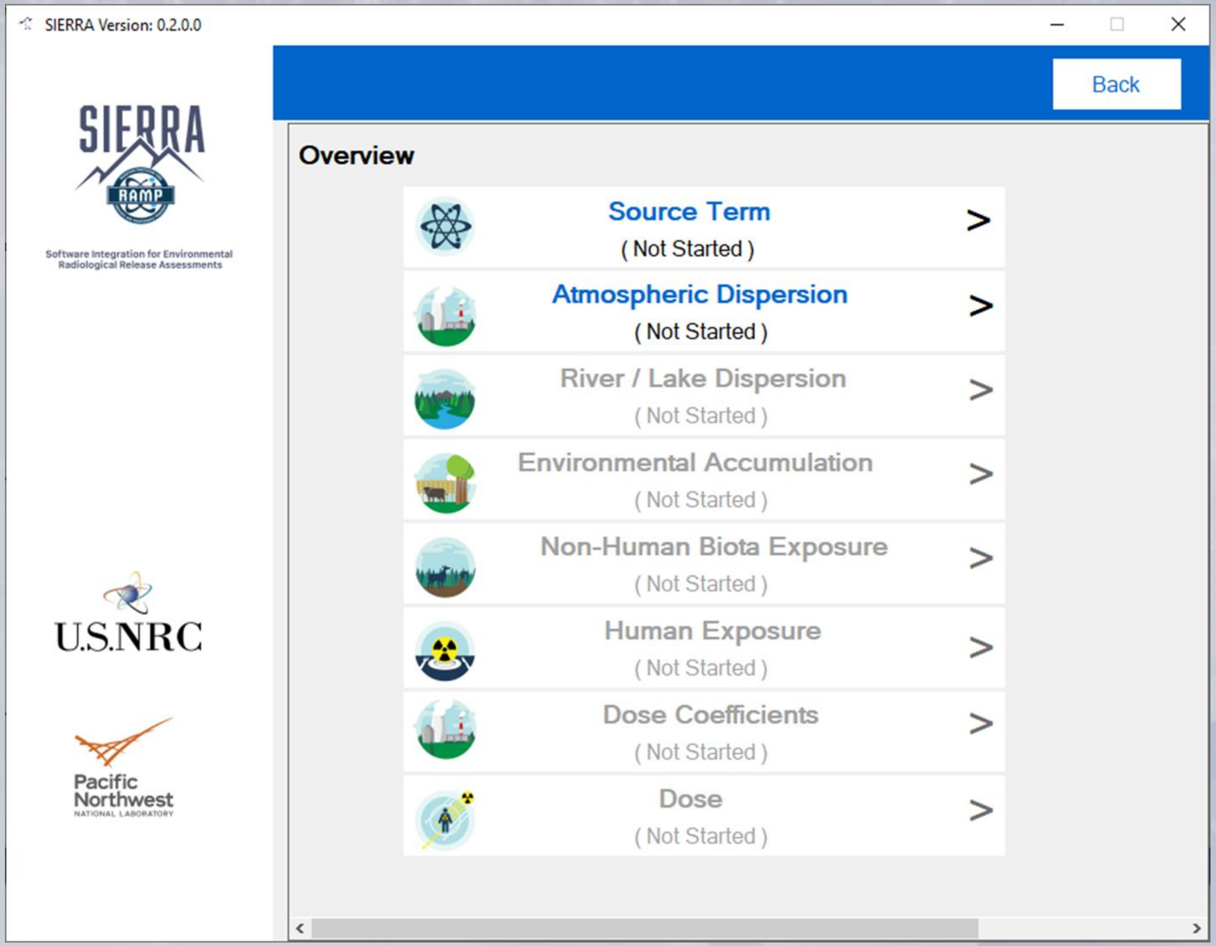
- Three Pillars:
  - Created consolidated engines/modules.
  - Developed a standardized data transfer schema.
  - Built a single user interface.



Task 1: Code Consolidation and Modernization (3/3)

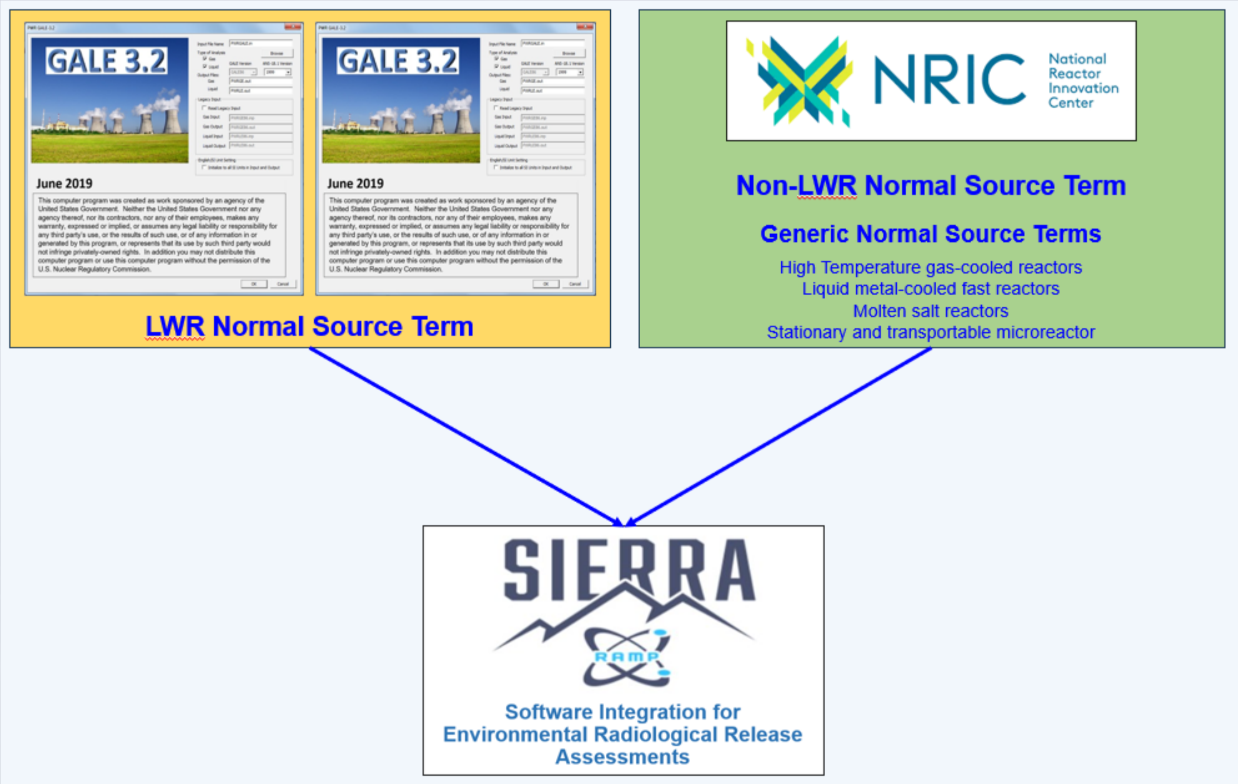
Phased Release of SIERRA

- Software under active development which aims to combine multiple licensing and siting codes into one easy to use package.
- Release of ATD Module of SIERRA at the end of September 2024.
- Currently have two efforts underway for SIERRA.
  - Atmospheric Dispersion Models (September 2024):
    - ARCON
    - PAVAN
    - XOQDOQ
  - Source Term:
    - GALE (Phase 1) – August 2024
    - Advanced reactors (Phase 2) – September 2025
  - Environmental Pathways (2026):
    - NRCDose3 (GASPAR & LADTAP)



Task 2: Improve characterization of Source Terms (1/4)

- Identify source terms inputs (i.e., radionuclide fuel inventories, reactor coolant inventories, plant design and operational data):
  - Phase 1: Incorporate LWR normal source terms.
  - Phase 2: Develop Non-LWR normal source terms.
  - Phase 3: Analyze Non-LWR design basis, severe accident and transportation source terms as applications of need arises.



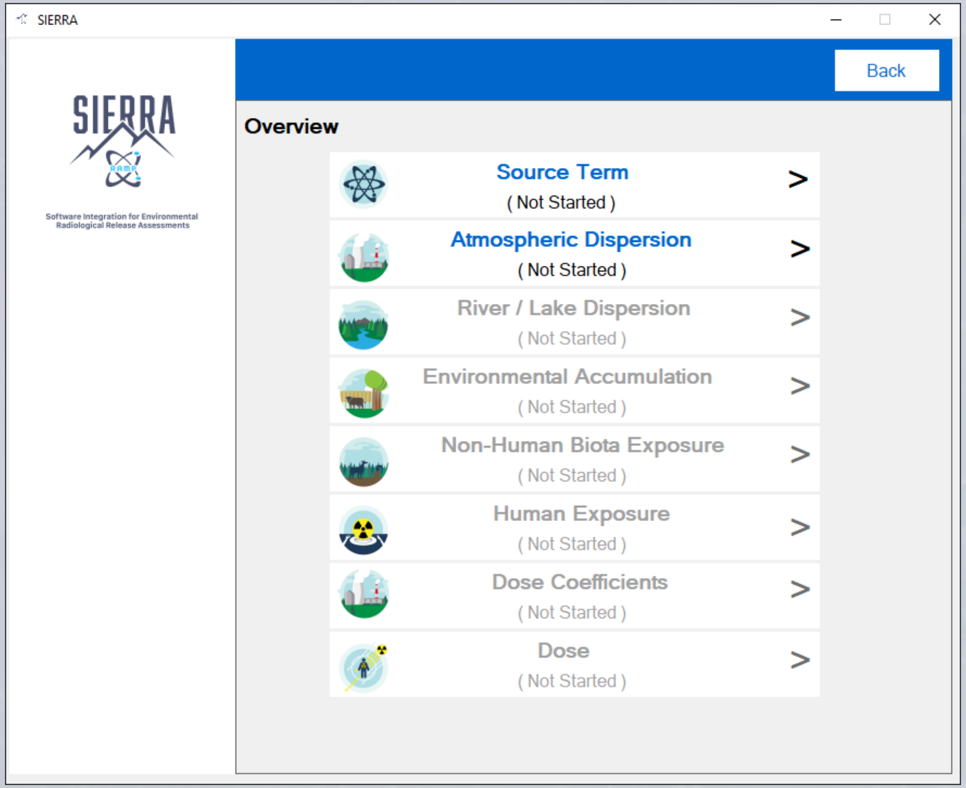
Task 2: Improve characterization of Source Terms (2/4)

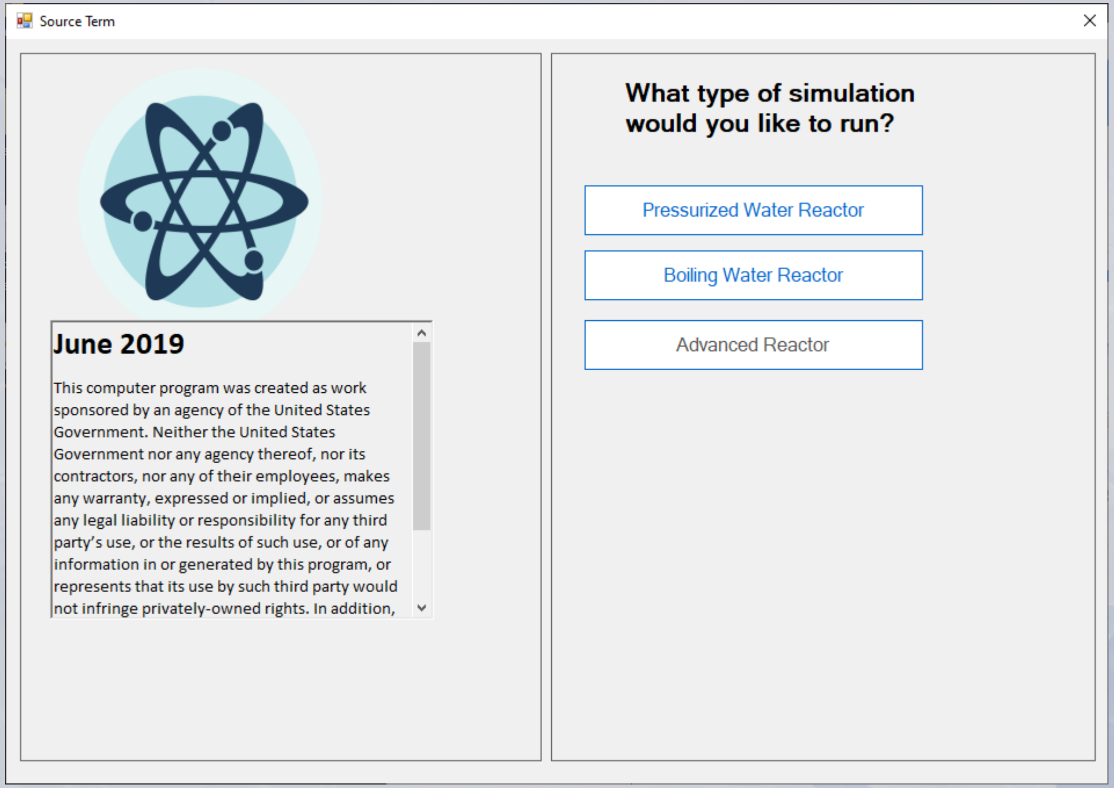
- Phase I - Input GALE code into SIERRA:

- Incorporating functionality of GALE (BWR and PWR) into the source term module.
- Status of GALE incorporation into SIERRA:
  - LWR normal source term module (Phase-1) to be available in August 2024.

Task 2: Improve characterization of Source Terms (3/4)

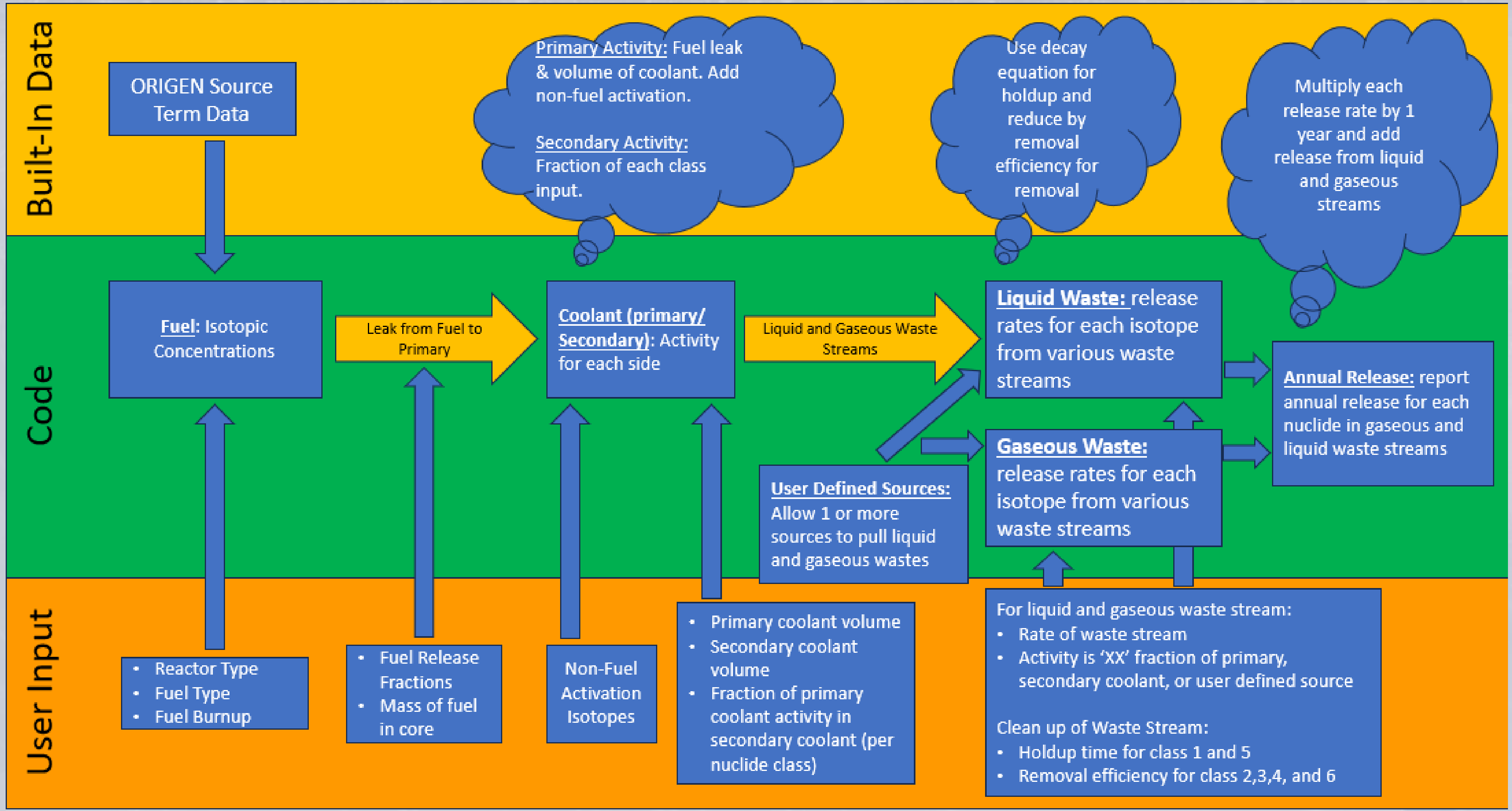
- GALE to SIERRA testing:
  - Numerical and Graphical User Interface (GUI) Verification and Validation (V&V) underway.





Task 2: Improve characterization of Source Terms (4/4)

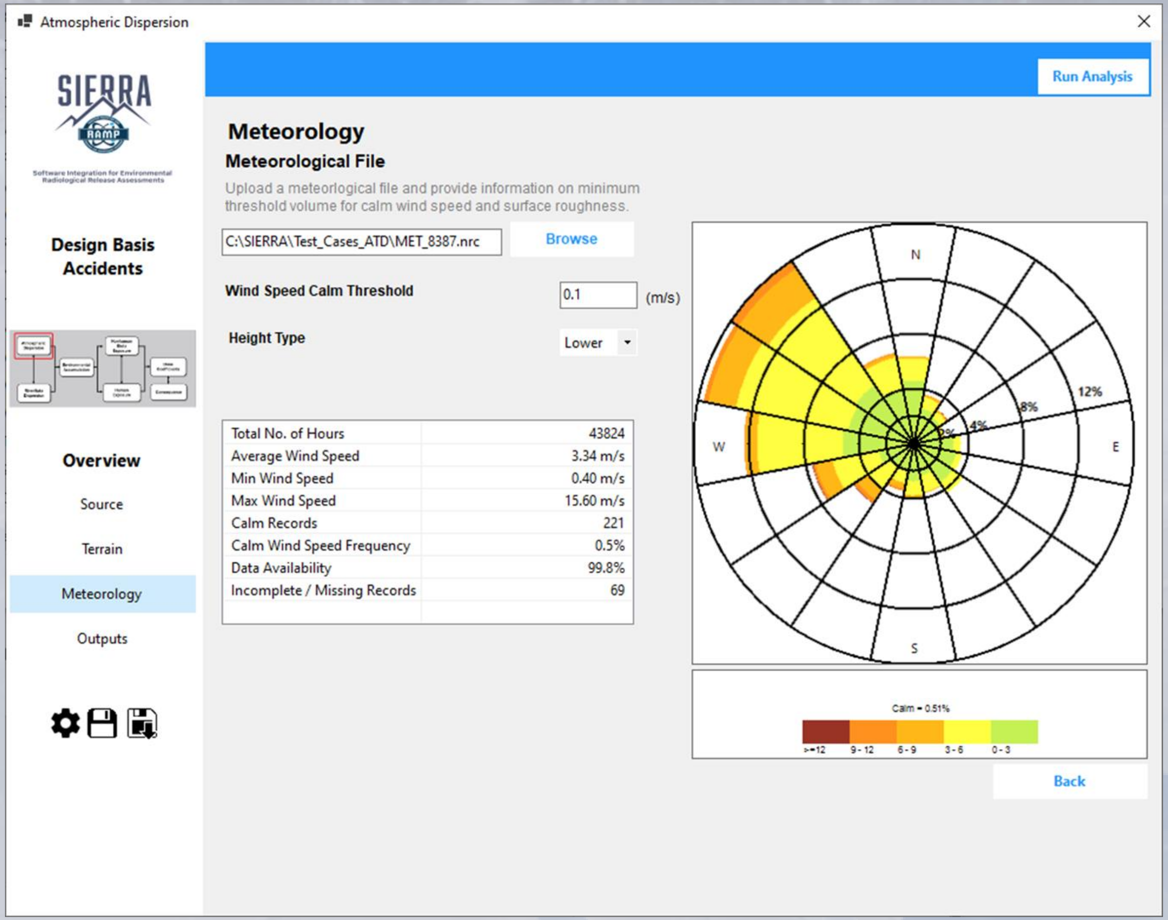
- Phase 2 - Input Generic Non-LWR Normal Source Terms into SIERRA:



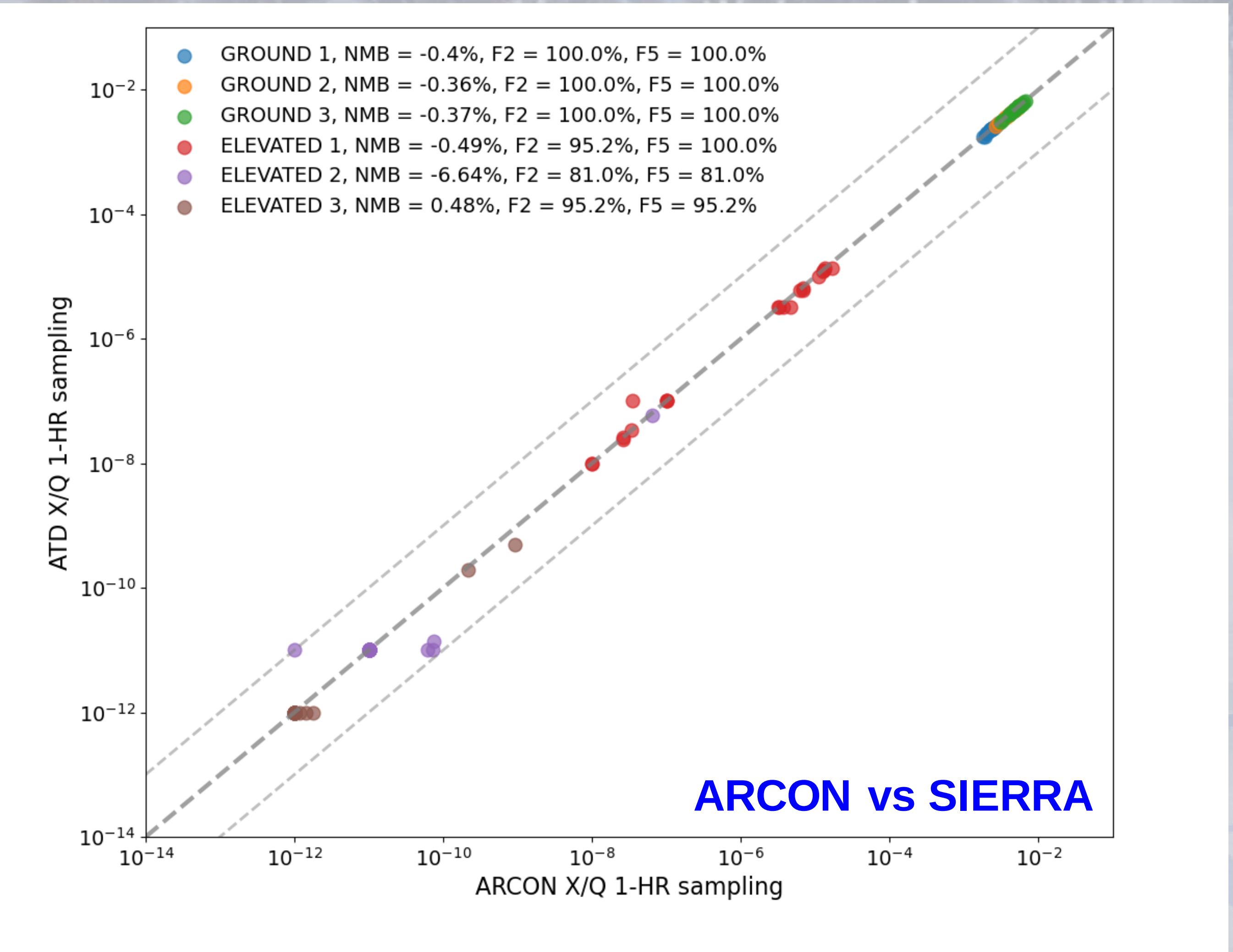
Task 3: Improve SIERRA ATD Models (1/4)

SIERRA ATD:

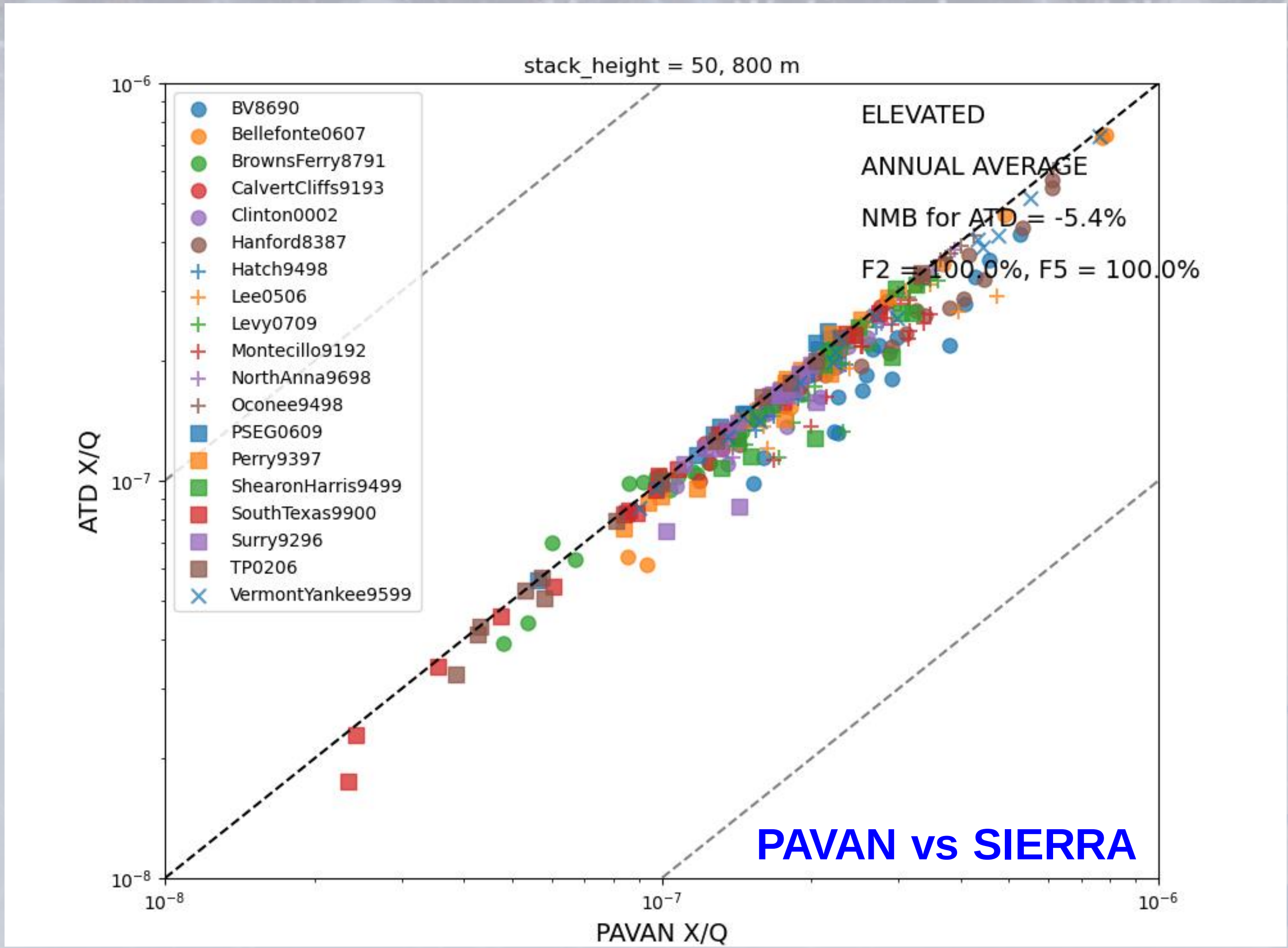
- Support a single user interface that allows users to access each of the codes (ARCON, PAVAN, XOQDOQ) in a relatively uniform manner.
- Facilitates future development to share data with other health physics codes in SIERRA.
- Allows users to estimate relative concentrations based on hourly meteorological data for all three codes, rather than use Joint Frequency Data.
- Written in a more modern version of FORTRAN.



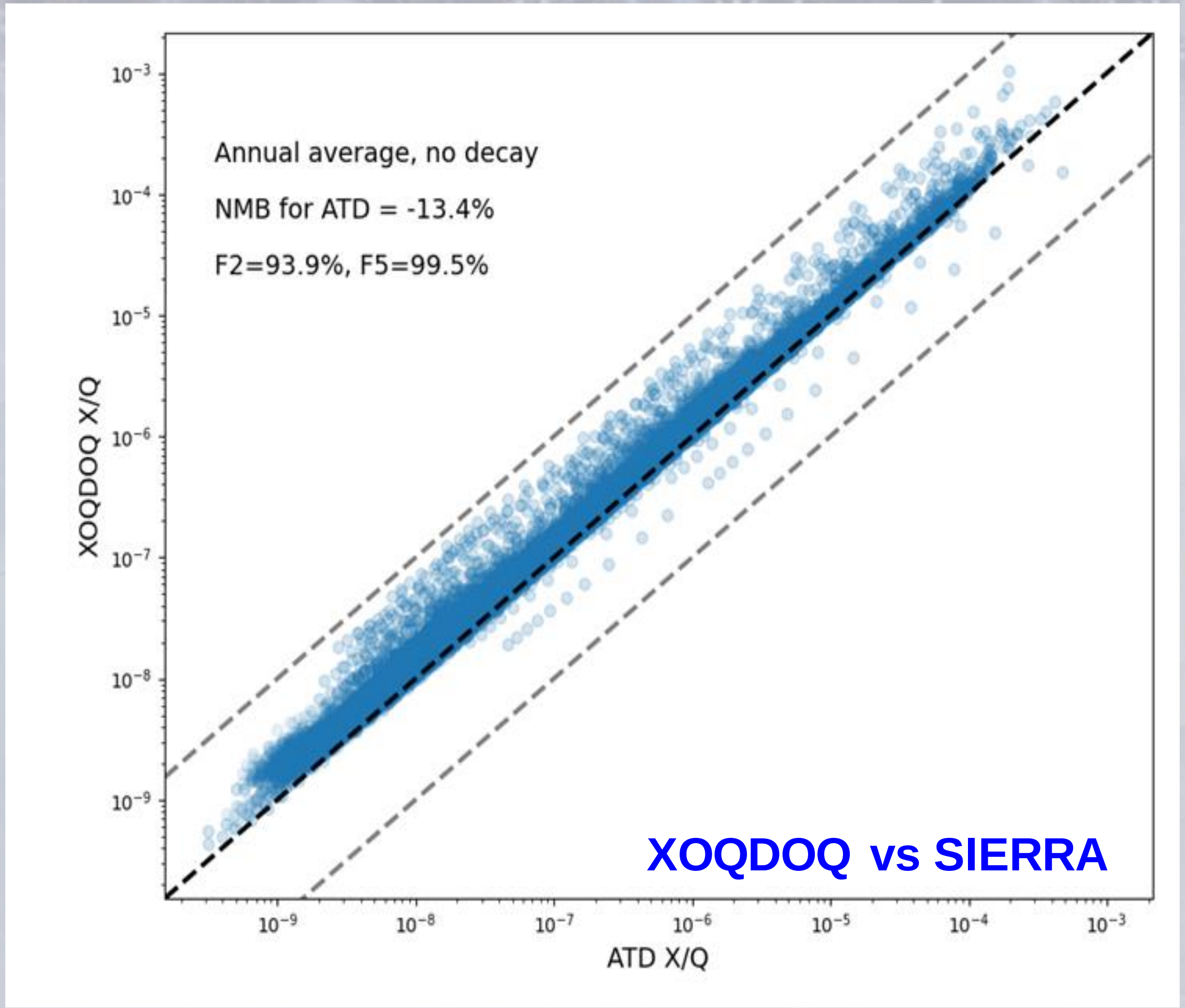
Task 3: Improve SIERRA ATD Models (2/4)



Task 3: Improve SIERRA ATD Models (3/4)

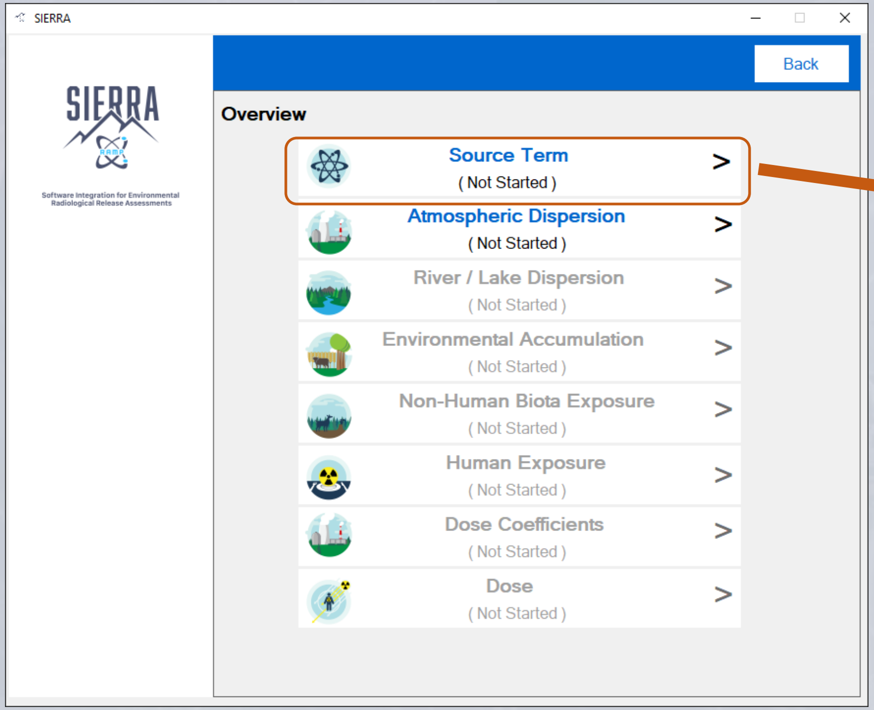


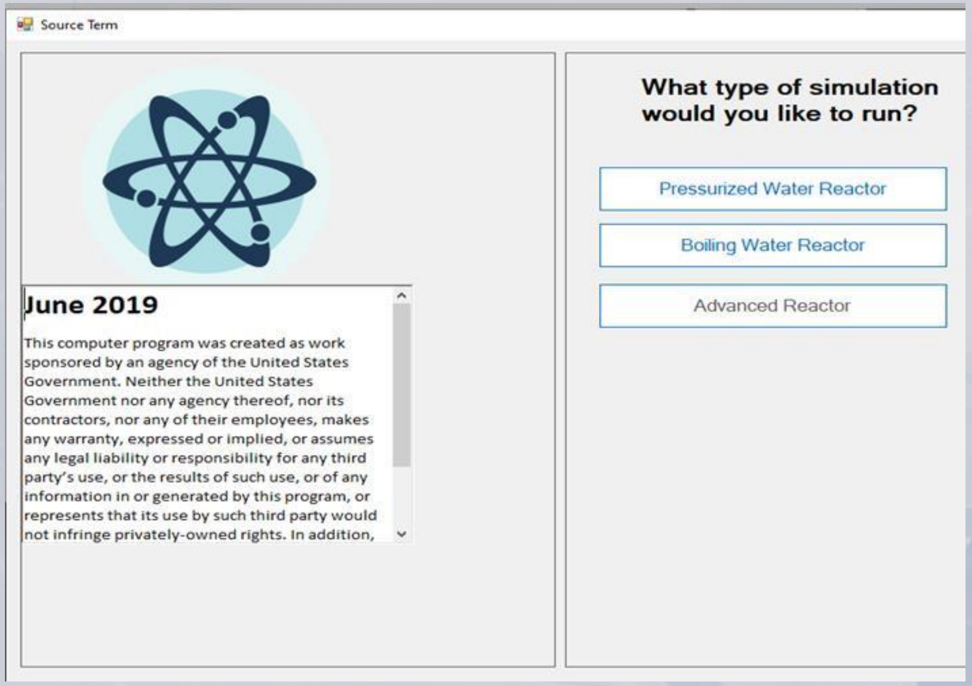
Task 3: Improve SIERRA ATD Models (4/4)



Next Steps

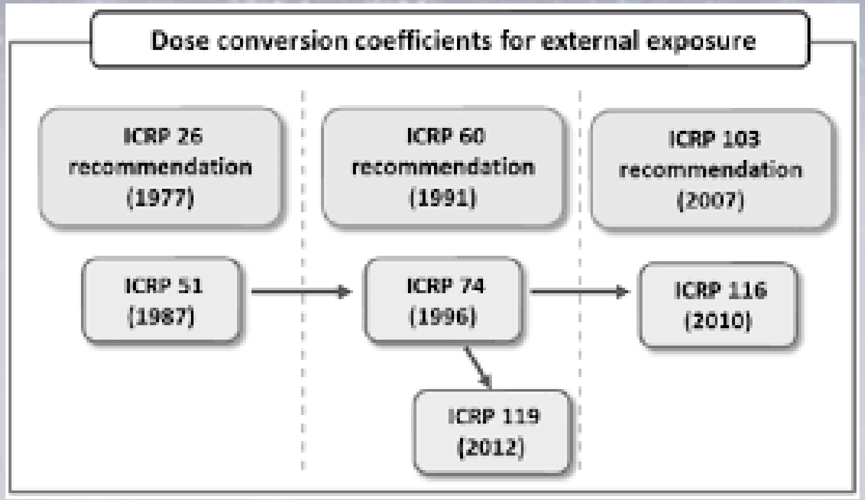
- Task 2: Improve characterization of Source Terms (Phases 2 & 3).
- Task 4: Update Dose Coefficient values.
- Task 5: Develop Environmental Pathway Models.

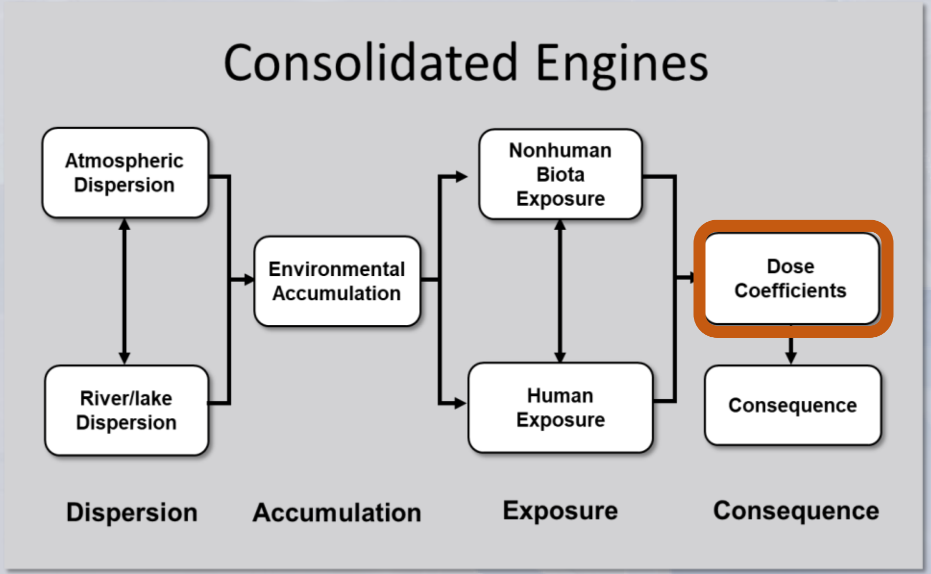




Task 4: Update Dose Coefficient Values

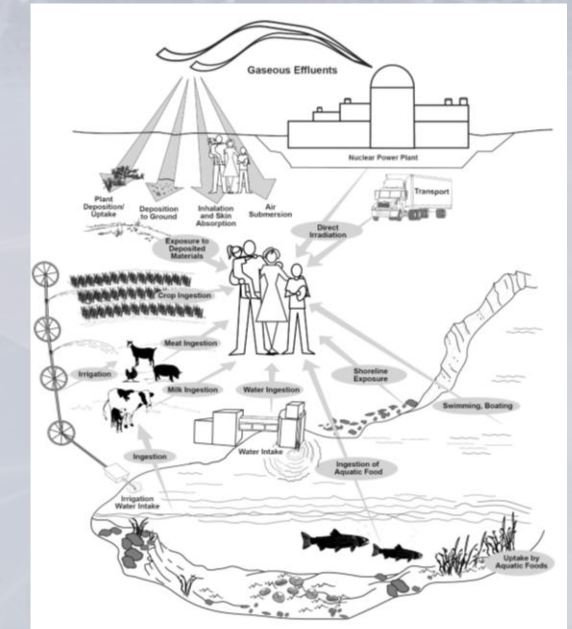
- This task involves:
  - Developing dosimetry modules/engines that have the flexibility to use different dose models and dose coefficient values.
  - Examining dose coefficient models with respect to aerosol particle size in addition to exploring the impact of tritium and carbon-14 biokinetics since these radionuclides may be in higher quantities in Non-LWRs.

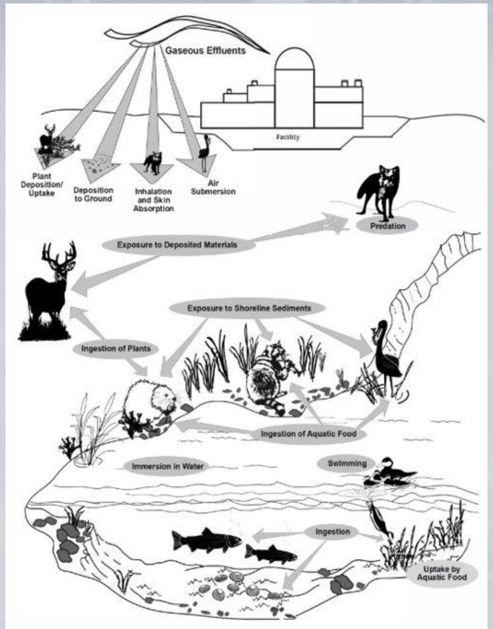




Task 5: Develop Environmental Pathway Models

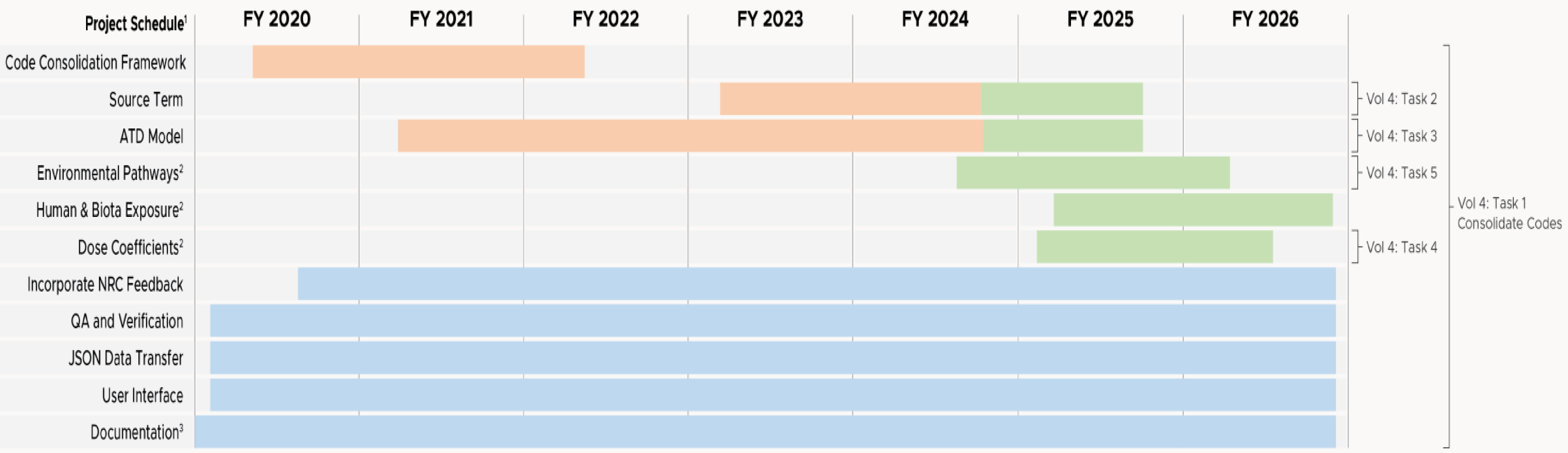
- Purpose:
  - Developing environmental transfer pathways and environmental accumulation.
- Current Status:
  - Exploring transferring NRCDose Computer Code into SIERRA.
  - Explore additional transfer model pathways for incorporation into SIERRA.
  - Explore modeling H-3 and carbon-14 accumulation in the environment.





NRCDose → Expanded Pathways → H-3, C-14, and Special Models

SIERRA Code Development Schedule



[1] These tasks generally align with 5 tasks described in ACRS Volume 4 briefing.

[2] This is the anticipated development schedule for the modules.

[3] Documentation–SQAP, Technical Basis Document, User Guide, Training Module.

Phase 1 Development
Phase 2 Development
Continuous Development

Summary of Code Readiness for Non-LWR Reviews

Current Readiness for Non-LWR Reviews:

- The ATD Computer Codes – Non-LWR review ready.
- The SNAP/RADTRAD Computer Code – Flexible to add DBA source term for Non-LWR reviews.
- The NRCDose3 Computer Code – Flexible to add environmental pathways and dose coefficients for Non-LWR reviews.

Next Steps for Readiness for Non-LWR Reviews:

- The SIERRA computer code:
  - ATD module – September 2024.
  - Normal Source Term – September 2025.
  - DBA Source Term (SNAP/RADTRAD) – September 2026.
  - Environmental Transport and Dose Coefficients – September 2026

Questions

Status Update on Computer Code and Model Development for Non-Light-Water Reactors

April 3, 2024

Fuel Cycle Analysis

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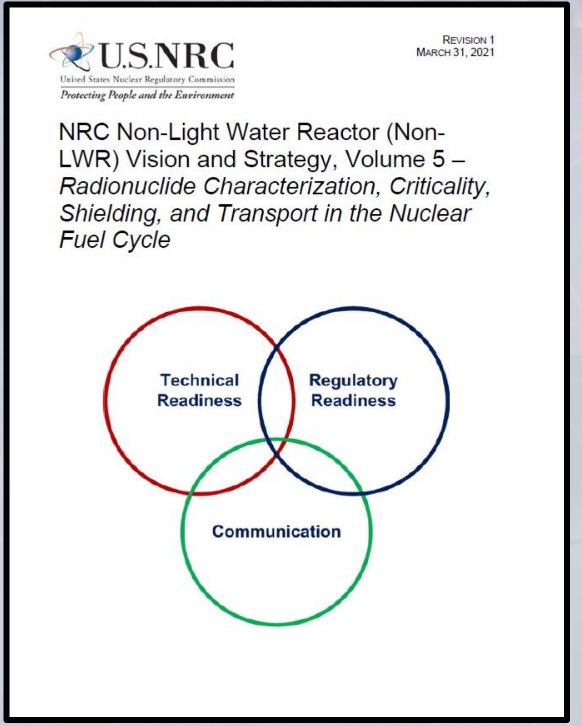
Reactor System Engineer

Shawn Campbell, Ph.D.
Reactor System Engineer

Andrew Bielen, Ph.D.
Senior Reactor System Engineer

Division of Systems Analysis
Office of Nuclear Regulatory Research

Fuel Cycle Analysis Objectives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REVISION 1
MARCH 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

ML21088A047

- Identify differences in potential non-LWR fuel cycles compared to LWR fuel cycle
- Identify capability gaps, in NRC's simulation capabilities (SCALE & MELCOR)
- Address any capability gaps through code development activities
- Assess, demonstrate, document through publicly available deliverables

Assess changes in the non-LWR fuel cycle & evaluate NRC's simulation capabilities for performing independent safety analyses

Fuel Cycle Analysis Approach

RES

- Based on publicly available information, develop models for stages of representative fuel cycles
  - Leverage the reference plants & reactor core designs from Volume 3

- Identify and select key accidents to model within SCALE & MELCOR, exercising key phenomena & models

- Develop and simulate representative SCALE & MELCOR models and evaluate
  - Identify areas where data gaps, high importance inputs, and areas to improve in our codes exist
  - SCALE – criticality, radionuclide inventory generation, decay heat, and shielding
  - MELCOR – radiological & non-radiological material & energy transport

NRC's computational capabilities will be demonstrated through public workshops and technical reports.

Fuel Cycle Analysis
Nuclear Fuel Cycle & Facility Accident Analysis

Types of Fuel Cycle Safety Analyses within Volume 5

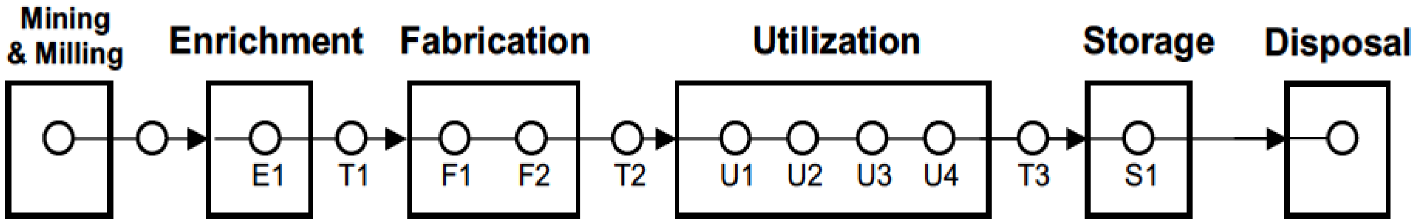
Criticality Safety	Radionuclide inventory & Decay heat generation	Radiation Shielding & Dose	Radiological material & energy release / transport	Non-radiological material & energy release / transport

Inadvertent nuclear criticality events

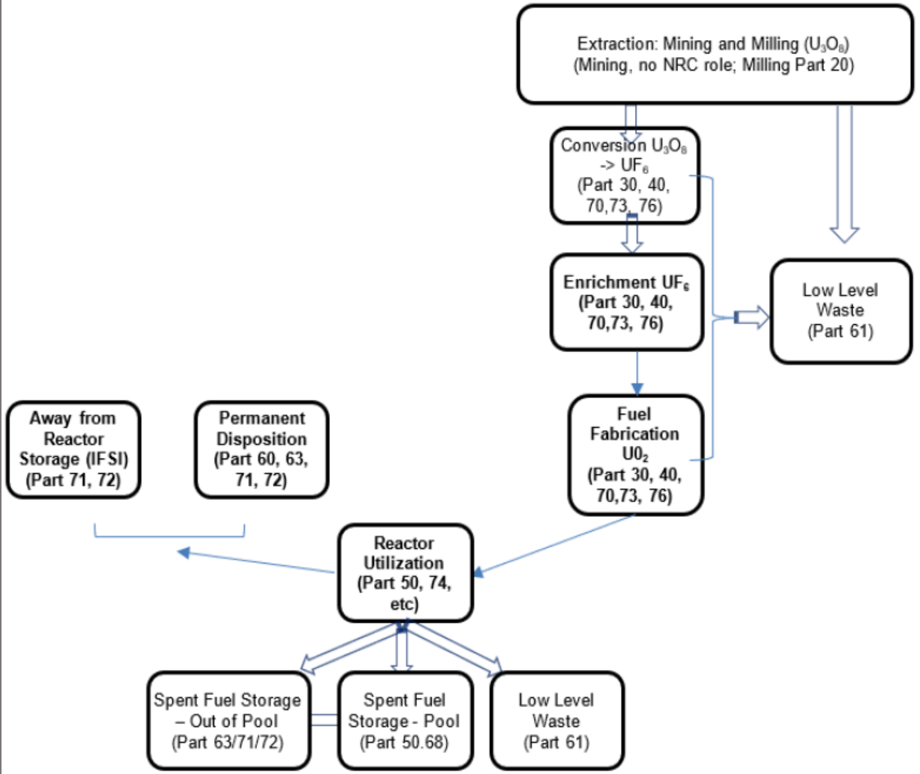
- Solution systems
- Powder systems
- Large storage arrays

NUREG/CR-6410 provides insights and methodology for performing fuel cycle safety analyses. Other references used include NUREG-1520, NUREG-2215, NUREG-2216.

Fuel Cycle Analysis
LWR Nuclear Fuel Cycle



E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of UO_2 fuel pellets
F2 – fabrication of LWR fuel assemblies
T2 – transportation of fresh fuel assemblies to the plant
U1 – fresh fuel staging and loading
U2 – power production
U3 – spent fuel pool/shuffle operations
U4 – on-site dry cask storage
T3 – transportation of spent fuel to off-site storage
S1 – off-site storage

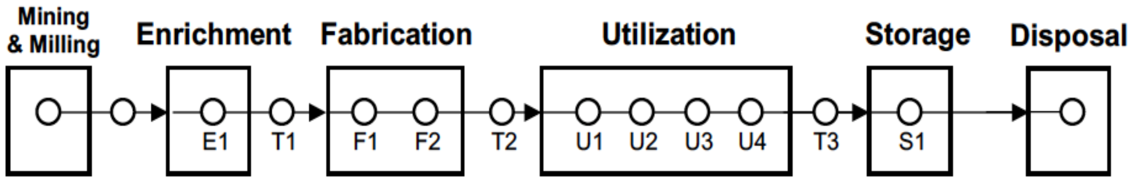


LWR open fuel cycle used as the starting point for developing each non-LWR fuel cycle.

Fuel Cycle Analysis Non-LWR Characteristics

	Enrich (%)	Fuel Form	Approx. BU (GWd/MTU)	Fuel Residence Time	Fuel Processing	Storage	Transportation
LWRs Baseline	< 5	Uranium Oxide	62	3-4 cycles (18 - 24 months per cycle)	No	Fresh / SNF storage on site or off - site	Fresh UF6 → 30B cylinders Fresh fuel → various packages Spent fuel → various packages and dry storage systems
HPR	< 20	Oxide Metal	Up to 10	Up to 7 years	No	TBD	TBD
SFR	< 20	Metal	Up to 300	TBD	No	TBD	TBD
HTGR	< 20	TRISO pebbles TRISO compacts	100 – 200	TBD	No	TBD	TBD
FHR	< 20	TRISO pebbles TRISO compacts	100 – 200	TBD	No	TBD	TBD
MSR	< 20	Liquid	TBD	2 – 3 years	Yes	TBD	TBD

Fuel Cycle Analysis
Non-LWR Nuclear Fuel Cycle



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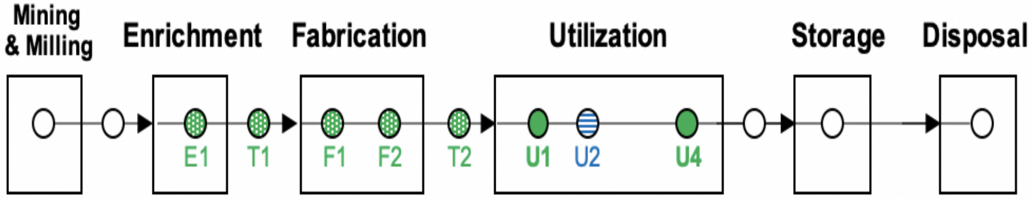
U4 – on-site dry cask storage
T3 – transportation of spent fuel to off-site storage
S1 – off-site storage



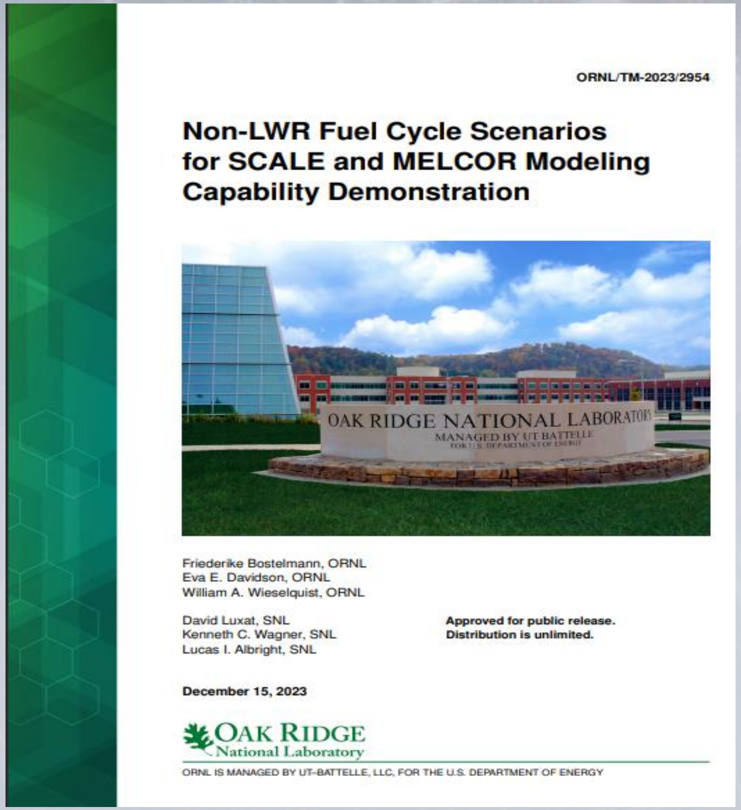
E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of TRISO fuel kernels
F2 – fabrication of fuel pebbles
T2 – transportation of fresh fuel pebbles to the plant
U1 – fresh fuel staging and loading
U2 – power production including online refueling
U4 – on-site discharged pebble storage

focus of this project report
covered with reference to other reports inside project
covered with reference to other activities outside project
not considered in this report

Fuel Cycle Stages Not Considered in Volume 5's Demonstration Project

Mining & Milling – No major changes envisioned from current methods.
Power Production – Executed under the Volume 3 umbrella.
Off-site Spent Fuel Storage & Transport – High degree of uncertainty for implementation.
Spent Fuel Final Disposal – High degree of uncertainty for implementation.

Fuel Cycle Analysis
Representative Fuel Cycle Designs


ORNL/TM-2023/2954

Non-LWR Fuel Cycle Scenarios for SCALE and MELCOR Modeling Capability Demonstration

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Lucas I. Albright, SNL

Approved for public release. Distribution is unlimited.

December 15, 2023

OAK RIDGE National Laboratory

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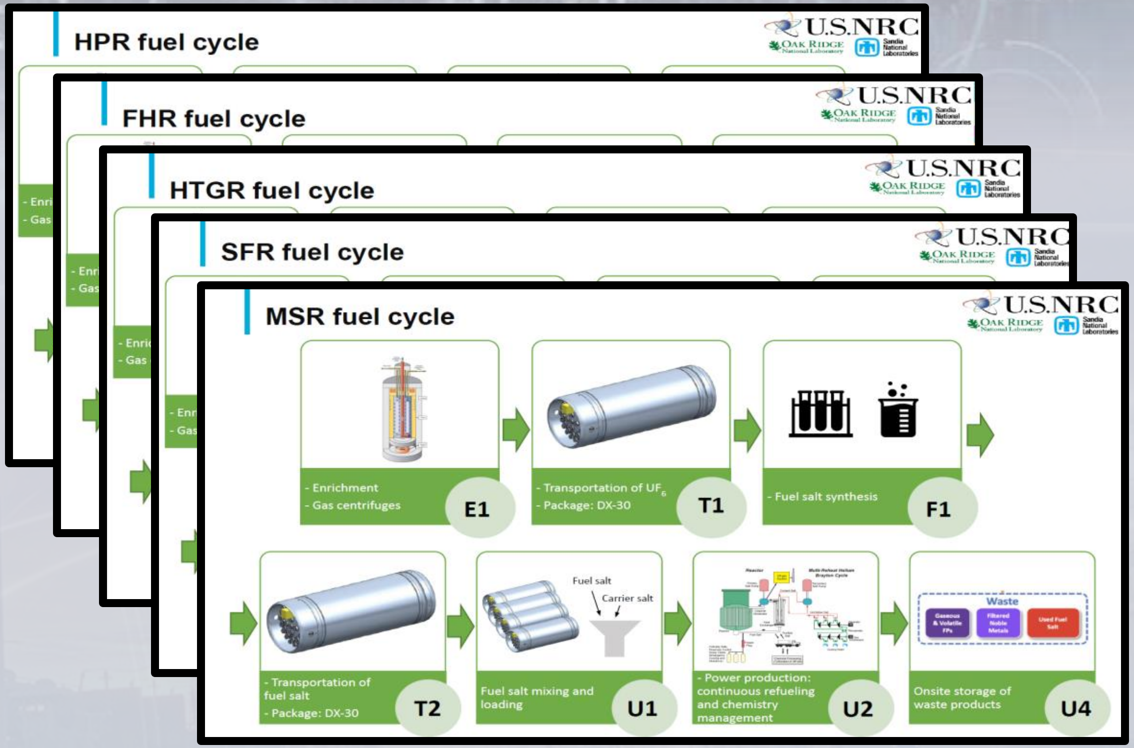


Developed five representative fuel cycle designs leveraging the Volume 3 reactor designs & identified potential accidents for the various stages of the fuel cycle.

Fuel Cycle Analysis
Types of Accidents Analyzed

Various Types of Fuel Facility Accidents

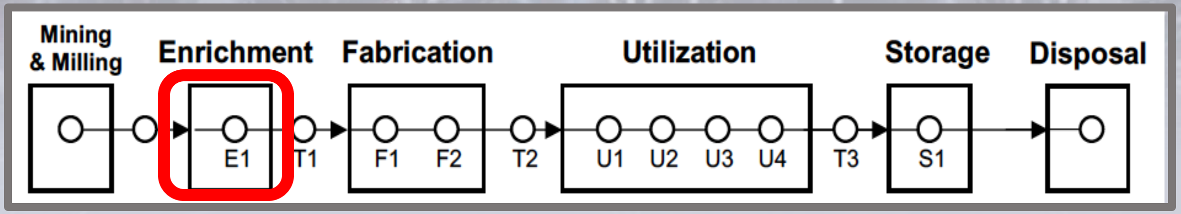
- Criticality Safety
- Radionuclide inventory & Decay heat generation
- Radiation Shielding & Dose
- Radiological material & energy release / transport
- Non-radiological material & energy release / transport

Water ingress during transportation of UF_6 in DN30-X

Dropped spent SFR fuel handling within containment during a seismic event

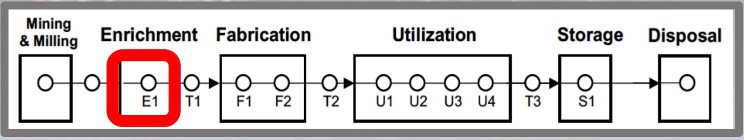
UF_6 cylinder is overfilled and heated, resulting in a rupture

Fuel Cycle Analysis Highlights - UF_6 Enrichment



Hazardous Material Identified	Potential Accidents
Inventory of hazardous chemicals identified (NH_3, F_2, HF, KOH, UF_6) UF_6 identified as the only source of dispersible radiological material in this fuel cycle stage.	Radiological Release ➢ UF_6 cylinder rupture (overfill/heated, damage/drop) Criticality Safety ➢ UF_6 criticality up to HALEU enrichment Non-radiological ➢ HF, NH_3, F_2 release (seismic / pipe rupture)

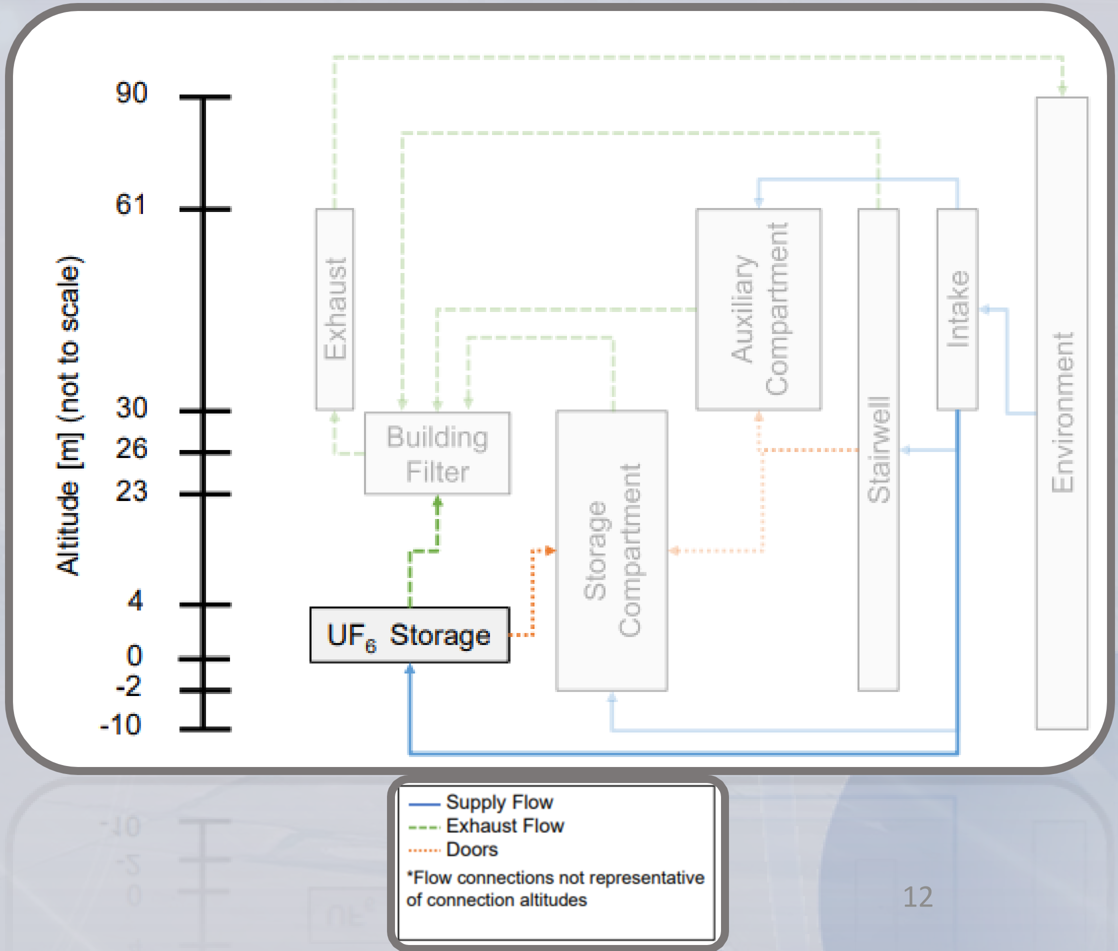
Fuel Cycle Analysis
UF_6 Cylinder Rupture- Chemical Hazard



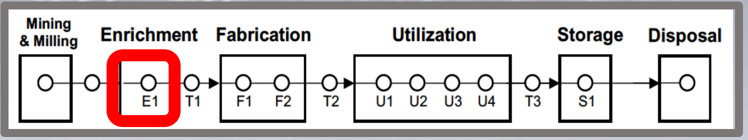
48Y cylinders may be used to store and transport UF_6. A 48Y is overfilled and heated, resulting in a tank rupture and rapid release of UF_6.

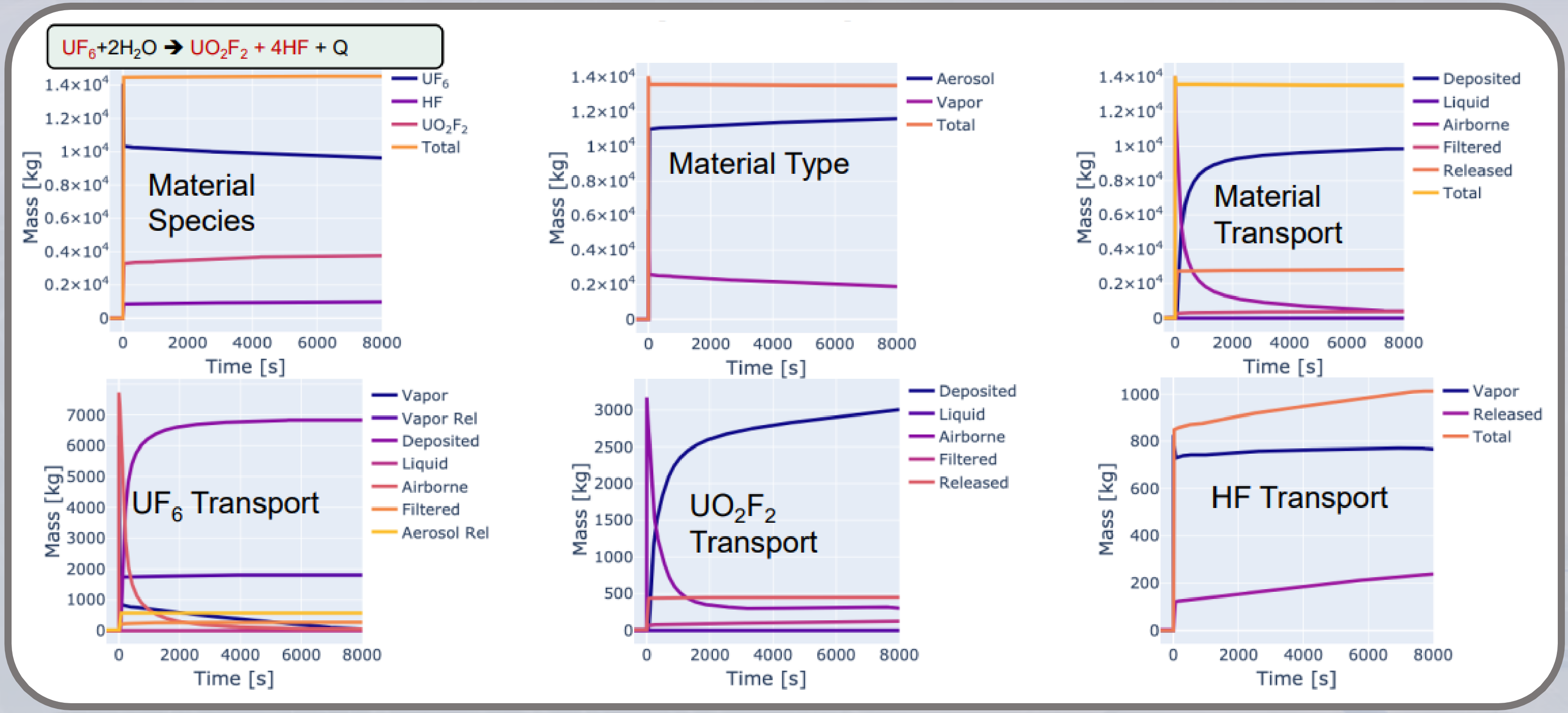
$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$

- MELCOR has robust capabilities and flexibility for aerosol and vapor release and transport modeling. It is leveraged here to model the release of UF_6 and its transport throughout the facility and into the environment.
- Modeling Assumptions
  - 14,000 kg of UF6 is stored within the 48Y, prior to release.
  - Instantaneous release.



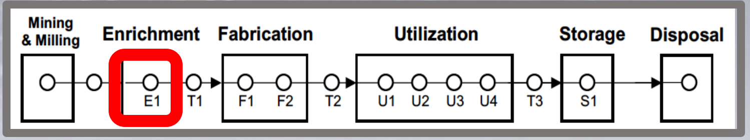
Fuel Cycle Analysis
UF_6 Cylinder Rupture- Chemical Hazard





Mass released primarily during initial rupture event, with minimal releases observed afterwards.
Masses are primarily aerosol and exhibit strong tendency to deposit on building structures.

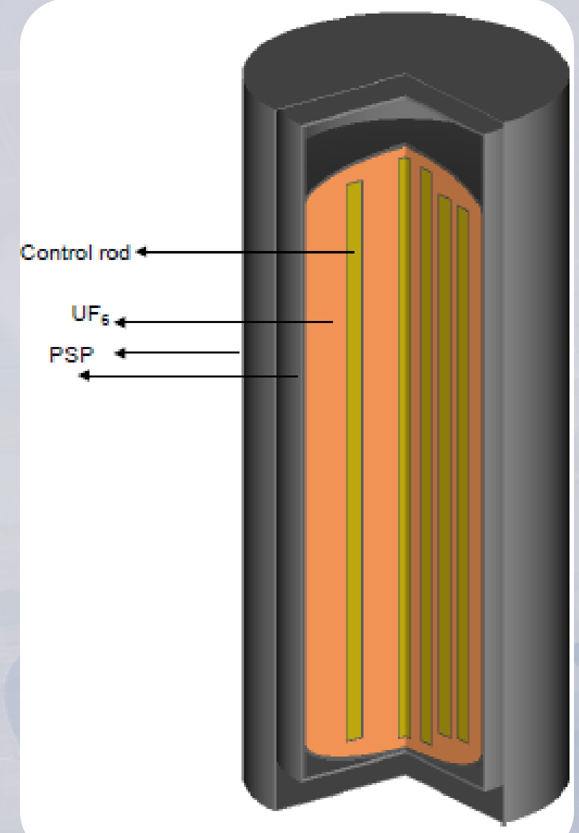
Fuel Cycle Analysis
UF_6 within DN30-X Package - Criticality Analyses



DN30-X is a transportation package designed with neutron poisons, for use with HALEU. Criticality safety analyses were performed for the following configurations:

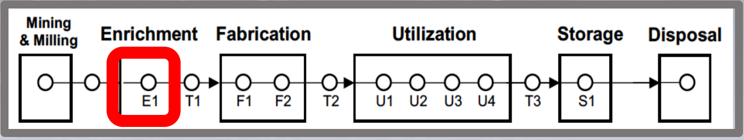
- Infinite hexagonal array; surrounded by air,
- Hexagonal array; surrounded by water, with no water between the outer and inner PSP
- Hexagonal array; surrounded by water, with water ingress between the outer and inner PSP

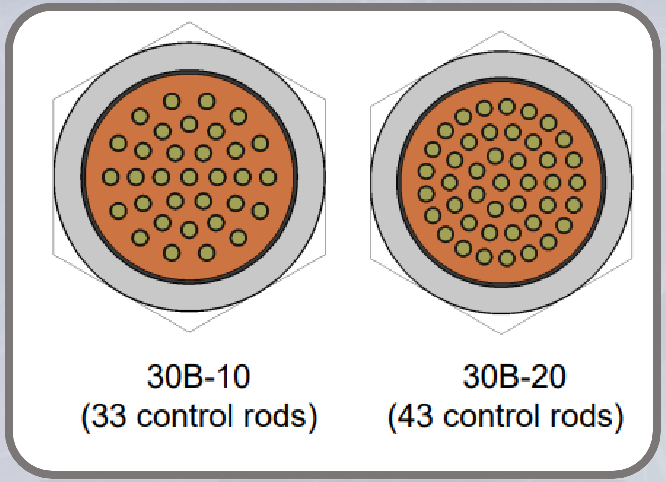
- SCALE/Shift used to perform the criticality safety analyses, using both ENDF/B-VII.1 & VIII.0.
  - Shift is SCALE's new high performance Monte Carlo neutron transport code.
- Modeling Assumptions
  - No thermal insulating foam modeled in the SCALE model
  - UF_6 density is assumed at 5.5 g/cm^3 with 0.5 wt.% HF impurities
  - Cylinders are 100% filled with UF_6 exceeding allowable mass limits of the cylinder
- Capabilities Demonstrated
  - SCALE's Shift for simulating HALEU enriched UF_6 (20 wt. % U-235) shipping packages in 3D



3D view of SCALE model

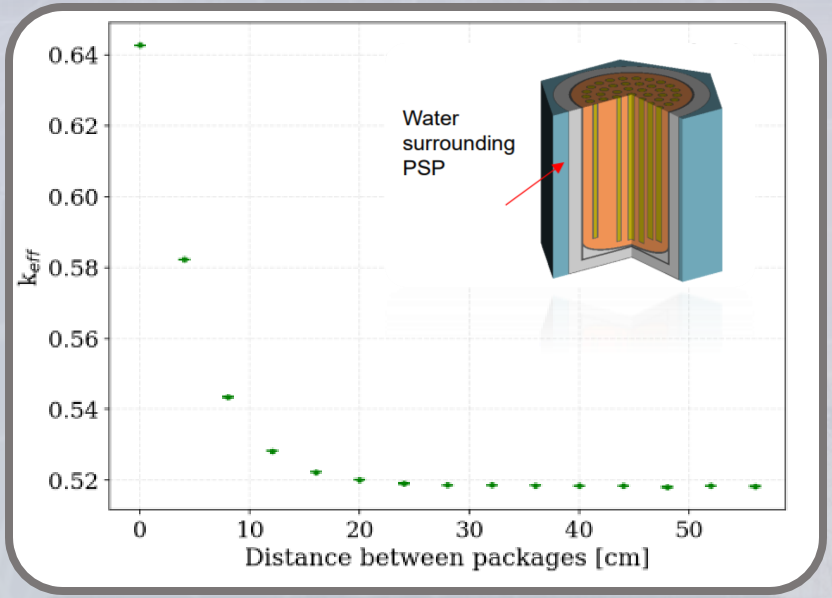
Fuel Cycle Analysis
UF_6 within DN30-X Package - Criticality Analyses



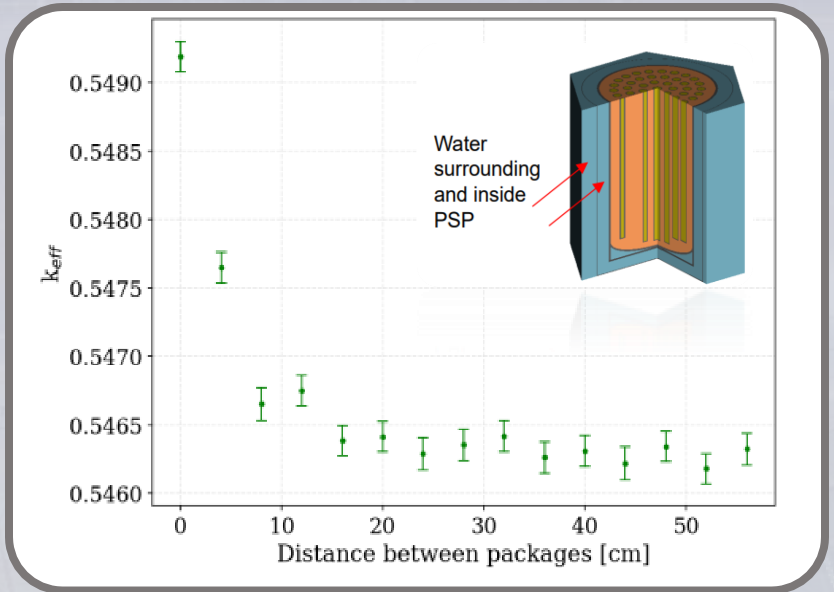


Nuclear Data Library	DN30-10	DN30-20
k_{eff} ENDF/B-VII.1 CE	0.58459 +/- 0.00011	0.77772 +/- 0.00011
k_{eff} ENDF/B-VIII.0 CE	0.58549 +/- 0.00010	0.77761 +/- 0.00011

Infinite hexagonal array of packages, touching on all sides, surrounded by air, with no water ingress.

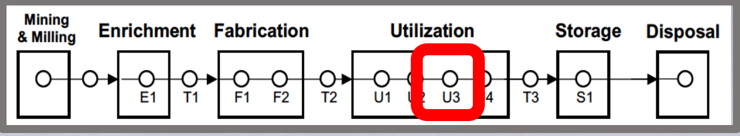


Array of packages, varied spacing, with water surrounding the PSP. No water ingress between outer and inner PSP boundary.



Array of packages, varied spacing, with water surrounding the PSP. Water ingress between outer and inner PSP boundary.

Fuel Cycle Analysis
SFR Fuel Handling Accident - Dose



During refueling operations, the refueling machine is used to perform fuel handling operations, such as moving spent fuel assembly in and out of the reactor core. A seismic event occurs causing the refueling machine to fail and drop a spent fuel assembly within the containment building.

- SCALE is used to determine the spent fuel nuclear inventory and perform the radiation dose estimates throughout the containment building. The radiation dose rate (radiative source term) is based upon an intact fuel assembly at various cooling periods.
- Modeling Assumptions
  - Spent fuel assembly is intact.
  - Containment building consists of a 1.2 cm thick steel liner, with reinforced concrete (1 m). Rebar-to-concrete mass ratio is 0.106.

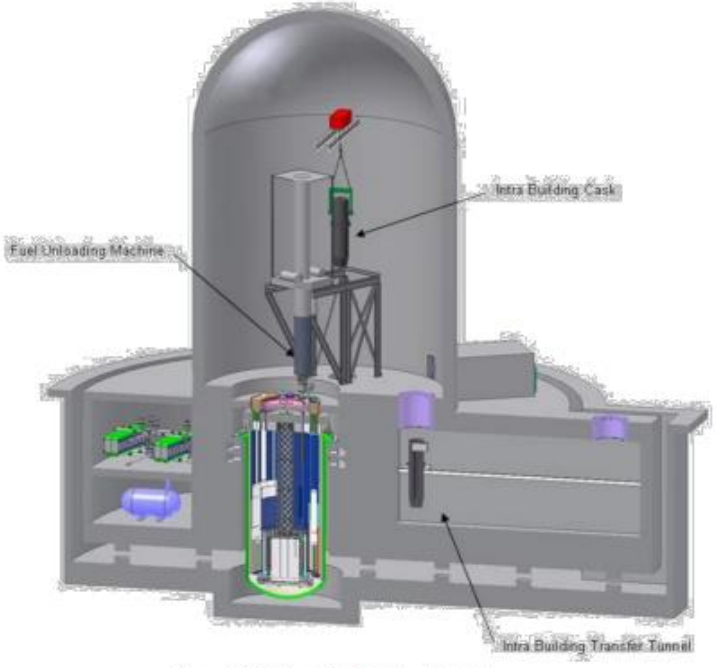
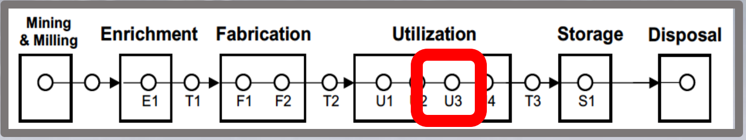
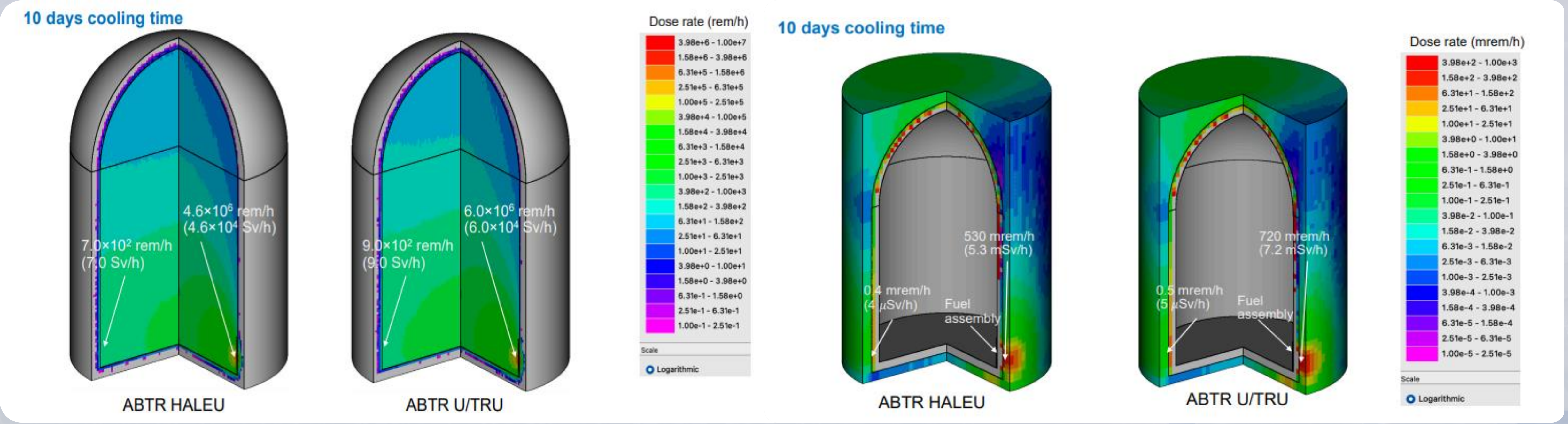


Figure II.7-1 Schematic of Fuel Handling System

ABTR reactor building

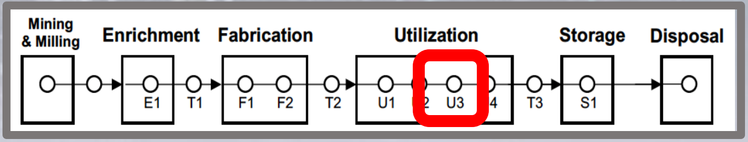
Fuel Cycle Analysis
SFR Fuel Handling Accident - Dose





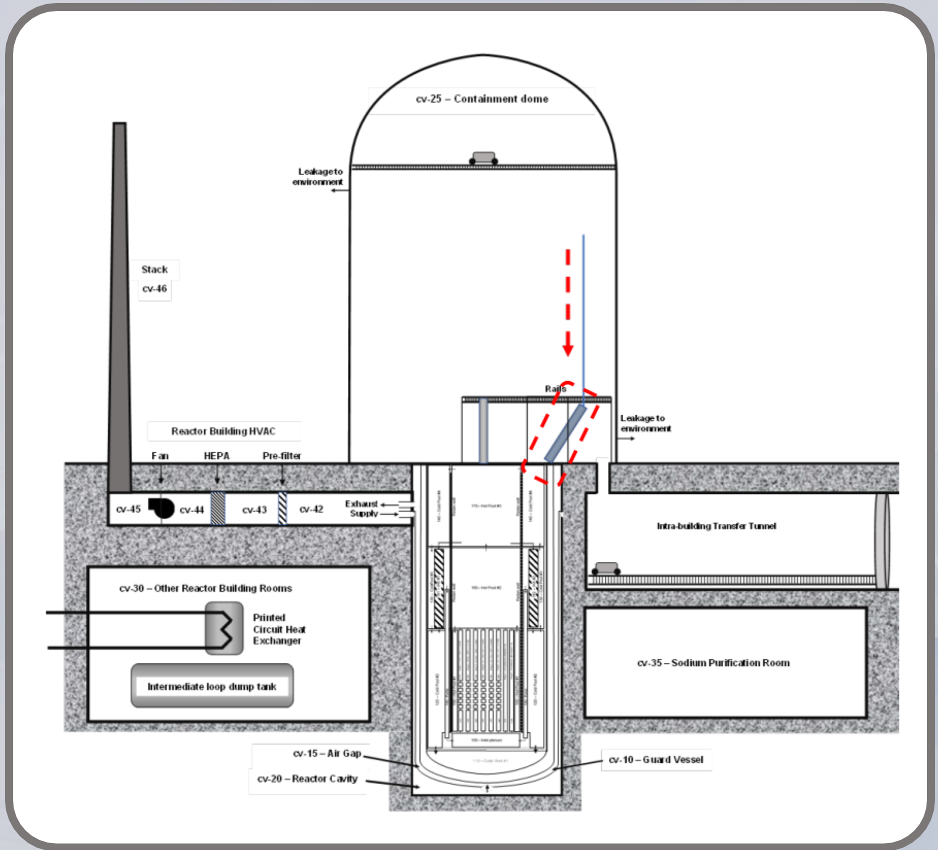
- Two cases analyzed; fuel cooled for 10 days & 7 reactor cycles.
    - 7 reactor cycles is the length of time a fuel assembly (FA) remains in the in-vessel storage.
- Neutron and gamma source terms determined for both ABTR HALEU & ABTR U/TRU fuel types.

Fuel Cycle Analysis
SFR Fuel Handling Accident –Material Transport

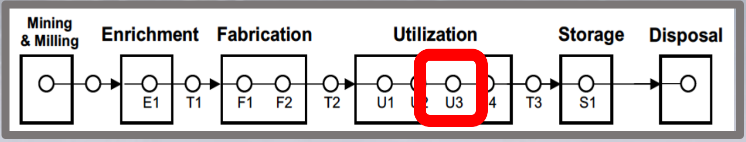


During refueling operations, the refueling machine is used to perform fuel handling operations, such as moving spent fuel assembly in and out of the reactor core. A seismic event occurs causing the refueling machine to fail and drop a spent fuel assembly loaded within a SNF cask within the containment building.

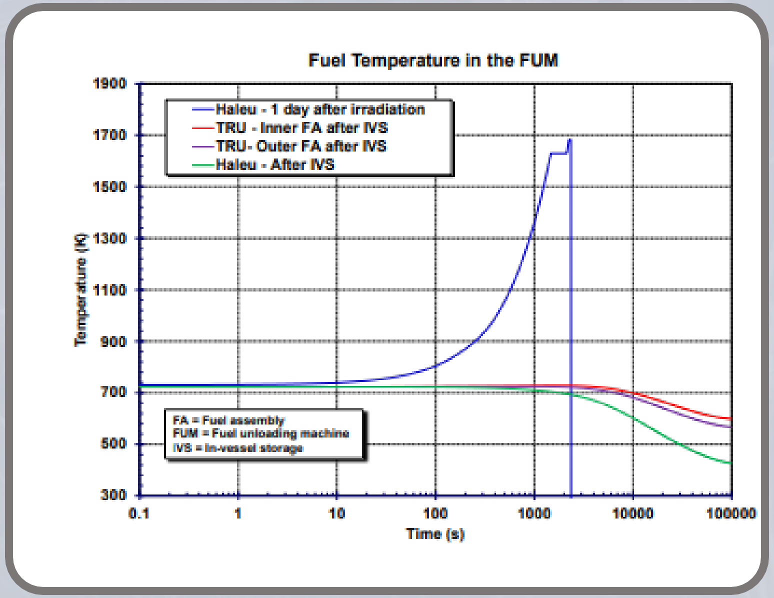
- MELCOR is used to model the fuel damage and radiological transport throughout the containment building. SCALE is used to provide the radionuclides for the HALEU spent fuel after in-vessel storage (7 cycles).
- Modeling Assumptions
  - No residual sodium in the cask.
  - All active cooling systems have failed.

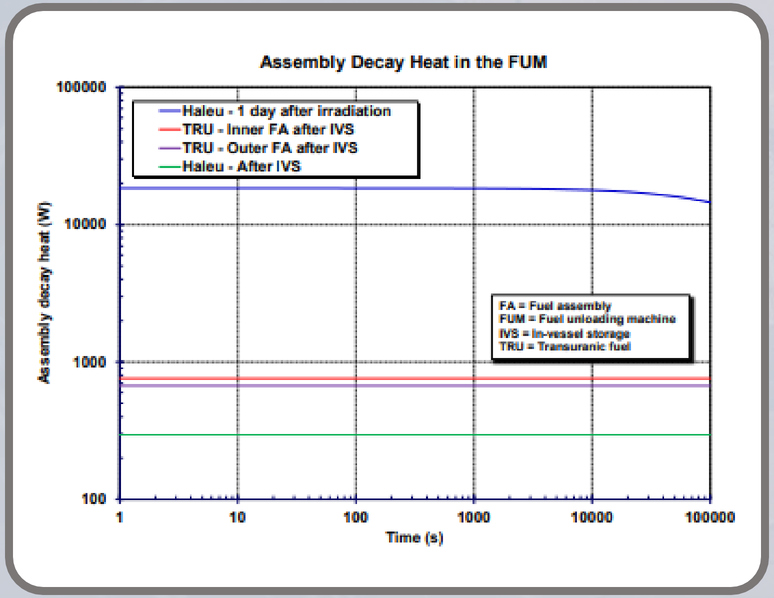


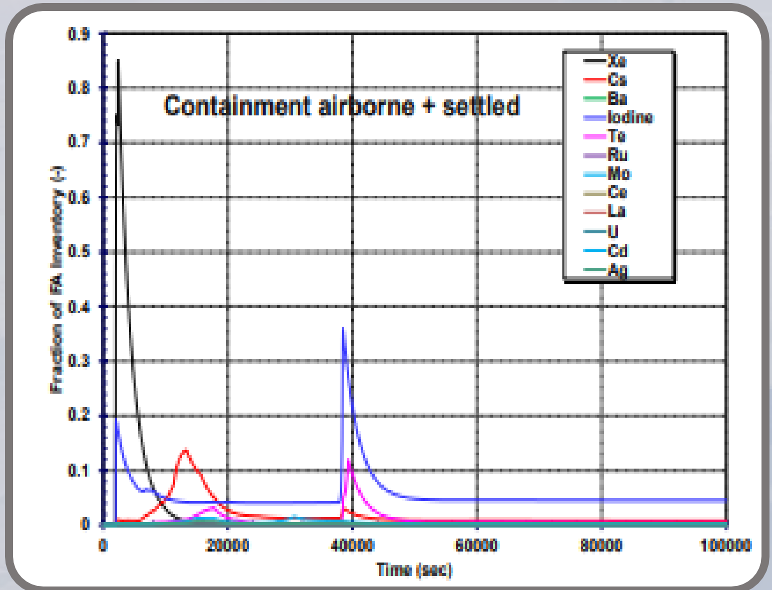
Fuel Cycle Analysis
SFR Fuel Handling Accident –Material Transport



- During removal from the reactor, FA are blown with argon gas to remove residual sodium.
- FAs with normal in-vessel storage cooling times remain intact within the failed fuel handling machine
- Accidental removal of a recently discharged FA would lead to fuel failures after 40 minutes.







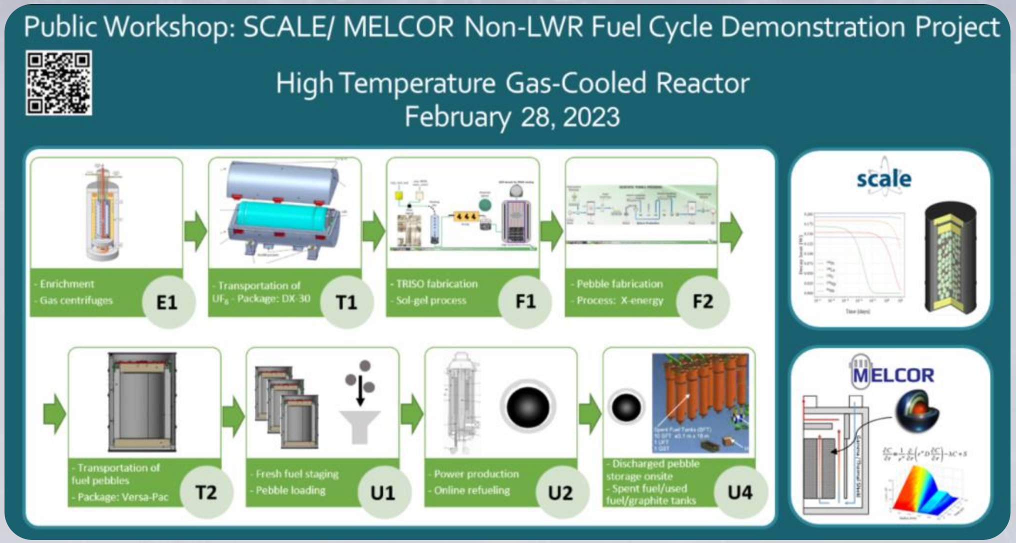
Fuel Cycle Analysis Public Workshops & Webpage

SCALE/MELCOR non-LWR fuel cycle demonstration project	

• High-temperature gas-cooled reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording EXIT
	February 28, 2023

• Sodium-cooled fast reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording
	September 20, 2023

• Non-LWR Fuel Cycle Scenarios for SCALE and MELCOR Modeling Capability Demonstration ◦ Report	December 15, 2023



Next planned workshop on the MSR fuel cycle will be Summer 2024

- Criticality during fuel salt conditioning
- Non-radiological release of beryllium during fuel salt conditioning
- Radiological release of fission products during a breach in the off-gas system
- Radiological release of tritium
- Dose analyses of the primary heat exchanger

Fuel Cycle Analysis
Key Highlights & Conclusions

RES

- Workshops and analyses have revealed some information gaps, for example:
  - No commercially-sized transportation packages for moving fresh pebbles.
  - Lack of public information for onsite fresh & spent fuel storage (pebbles, SFR fuel, etc.).

It is not envisioned this will challenge SCALE/MELCOR since no new models are required.

- The need for validation data (criticality safety benchmarking) has been identified, especially for TRISO based systems.
  - New collaboration between DOE and NRC for the Development of Criticality Safety Benchmarking Data for HALEU Fuel Cycle and Transportation (DNCSH)
  - Goal is to produce high-quality publicly available benchmarking experiments, nuclear data, and evaluations applicable to a wide range of HALEU systems.

SCALE & MELCOR demonstration workshops have shown NRC is ready to support fuel cycle analyses

Fuel Cycle Analysis Next Steps

RES

- Code development activities ongoing
  - MELCOR/ORIGEN Integration for MSR analyses
  - Capability to model multiple working fluids in the same MELCOR plant model
  - Addition of limited unstructured mesh capability to allow analysis of complex, arbitrary geometries of fissile material (e.g., fractured / damaged TRISO pebbles) in SCALE.
  - Improved modeling capabilities in SCALE to control-blades within pebble bed systems.
- Maintain awareness of industry priorities
- Training and knowledge management

Status Update on Computer Code and Model Development for Non-Light-Water Reactors

April 3, 2024

Conclusions

Kimberly A. Webber, Ph.D.

Director, Division of Systems Analysis

Office of Nuclear Regulatory Research

Conclusions

RES

Completed

- Non-LWR Code Development Reports
- Reference Plant Models
- SCALE/MELCOR Demonstration Public Workshops
- MACCS assessments and updates
- Code Assessment Reports for Metallic and TRISO Fuels
- Training on BlueCRAB Codes

Next steps

- New and Updates to Existing Reference Plant Models
- Verification and Validation (V&V) Report for Systems Analysis
- Assessment of MACCS capabilities to model physiochemical transformations during atmospheric dispersion
- Development/consolidation of Radiation Protection Codes for non-LWR analysis
- Fuel Cycle Demonstration Project Public Workshop for Molten Salt Reactor

Information Request from NuScale ACRS Subcommittee Meeting – March 19, 2024

Item 1 – Chapter 2: Concerning Hydrometeorological Reports

The staff does not consider it to be necessary that the NuScale Standard Design Approval Application (SDAA) include a statement requiring a site-specific precipitation study with the use of the most contemporary NOAA HMR report (Hydrometeorological Report) (or equivalent) to ensure climate change is accounted for in the meteorological sections impacting the design. SDAA COL Item 2.0-1 directs future applicants referencing the NuScale US460 design to demonstrate that the site-specific characteristics are bounded by the site parameters specified in SDAA Table 2.0-1. If those values are not bounded, then the applicant will demonstrate the acceptability of the site-specific values. If new precipitation studies are available at the time of the application, then the applicant should follow the guidance provided in DG-1290 (soon to be Revision 3 of RG 1.59), which states that "PMP [Probable Maximum Precipitation] values provided by HMRs should be evaluated in light of precipitation events that have occurred in the region since the HMRs were published. … If an alternative source other than an HMR prepared by the National Weather Service is used for the PMP estimate, the basis for the specific PMP value used needs to be explained. Considerations on an acceptable approach to the estimation of a site-specific PMP as an alternative to an HMR-based estimate can be found in NUREG/KM-0015." Current NOAA HMRs provide conservative extreme precipitation estimates and are accepted by both the NRC and the nuclear industry. When new data from NOAA or the National Academy of Sciences is available, the NRC will review the data and update the guidance as appropriate. Any applicant referencing the NuScale US460 design must demonstrate that the site is able to be protected against extreme precipitation and is bounded by the site parameters identified in SDAA Table 2.0-1.

Item 2 – Chapter 13: Regarding COL Item 13.5-7 Consistency in the EOPs

Roadmap for Plant Procedures

- Regulatory Requirements
- Standard Review Plan (SRP) guidance
- Expectations for SDAA and COLA submittals
- The Procedure Generation Package
- Construction Inspection Program

Regulatory Requirements

The NRC requires COL holders to have procedure programs. There is no regulatory requirement for EOPs to be consistent at sites with the same standard design.

- 10 CFR 52.79 Contents of applications, technical information in final safety analysis report (applicable to Combined Licenses)

  (27) Managerial and administrative controls to be used to assure safe operation. Appendix B to 10 CFR part 50 sets forth the requirements for these controls for nuclear power plants. The information on the controls to be used for a nuclear power plant shall include a discussion of how the applicable requirements of appendix B to 10 CFR part 50 will be satisfied;

  (29) (i) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components;

  (ii) Plans for coping with emergencies, other than the plans required by § 52.79(a)(21);

- 10 CFR Part 50, Appendix B, Criteria V and VI, establish criteria for development, approval, and control of procedures for all activities affecting quality.

- 10 CFR 52.137 Contents of applications; technical information, FSAR information (applicable to Standard Design Approvals):

  (a)(8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii),(f)(2)(ix), and (f)(3)(v) of 10 CFR 50.34(f);

- 10 CFR 50.34 (f)(2)(ii): Contents of Applications; technical information, additional TMI-related requirements:

  Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts.

NRC Staff Review Procedures

The NRC staff reviews **procedure programs** for normal operation, abnormal and emergency operation, testing and maintenance, and administrative controls.

- NUREG-0800, Standard Review Plan, Chapter 13 Conduct of Operations, 13.5.2.1 Operating and Emergency Operating Procedures, Revision 2, March 2007

- NUREG-0711, HFE Program Review, Section 9, Procedure Development:

  Procedures are integral to an overall HFE program and should be developed and implemented using accepted HFE principles

  The NRC reviews procedures to confirm that the applicant's procedure development program incorporates HFE principles and criteria

Expectations for SDAA and COLA submittals

At the **SDA application** stage, the staff reviews COL action items for procedures.

- Plant procedures include administrative procedures, operating procedures, emergency operating procedures as well as maintenance and other procedures for safety-related activities. The COL applicant is responsible for these types of procedures
- The staff's review is focused on the evaluation of COL action items pertaining to procedures
- The staff reviewed the COL information items in NuScale SDAA Section, 13.5, "Plant Procedures" for a COL to provide procedure descriptions and information about procedure program development and implementation.
- For example: COL Item 13.5-5 addresses EOPs:

  An applicant that references the NuScale Power Plant US460 standard design will provide a plan for the development, implementation, and control of emergency operating procedures, including preliminary schedules for preparation and target dates for completion.

  Additionally, the applicant will identify the group within the operating organization responsible for maintaining these procedures.

COL applicants or COL holders are required to develop procedures that are plant-specific.

- The COL application:

- May be received prior to development of detailed procedures
- Should contain a target date for completion of procedures
- Should describe the different classifications of procedures
- Should describe applicant's programs for developing procedures

- Procedures may be submitted after a COL is issued
    - Technical guidelines for developing EOPs are submitted as part of the Procedure Generation Package (PGP) at least 3 months before operator training on EOPs begins
    - Operating procedures need to be established, implemented and maintained at least 6 months prior to fuel load to allow for operator licensing examinations
    - Procedures are inspected as part of the Construction Inspection Program

The Procedure Generation Package

Information about EOP development and implementation is submitted for NRC staff review via the Procedures Generation Package (PGP)

- PGP must be submitted for NRC staff review no later than 3 months before formal operator training on EOPs begins
- PGP contains the following (from SRP Chapter 13.5.2.1):
    1. Plant specific technical guidelines (P-STGs) (these may or may not reference Generic Technical Guidelines)
    2. Plant specific writer's guide that details the methods to be used by the applicant in preparing EOPs based on P-STGs
    3. A description of the verification and validation program for EOPs
    4. A description of the program for training operators on EOPs
- The P-STGs must be derived from approved analyses of transients and accidents so that EOPs will be based on acceptable technical guidelines

Construction Inspection Program

The NRC staff verifies the technical adequacy of a COL holder's operating procedures through the **Construction Inspection Program**

Inspection Procedures (IPs) used by the staff during plant construction:

- IP 42401, Part 52, Plant Procedures
- IP 42453, Part 52, Operating Procedures Inspection
- IP 42454, Part 52, Emergency Procedures