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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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FUTURE PLANT DESIGNS SUBCOMMITTEE

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TUESDAY

DECEMBER 1, 2020

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The Subcommittee met via Video-teleconference, at 9:30 a.m. EST, Dennis Bley, Chairman, presiding.

COMMITTEE MEMBERS:

DENNIS BLEY, Chairman

RONALD G. BALLINGER, Member

CHARLES H. BROWN, JR. Member

VESNA B. DIMITRIJEVIC, Member

WALTER L. KIRCHNER, Member-at-large

JOSE MARCH-LEUBA, Member

DAVID A. PETTI, Member

JOY L. REMPE, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

MICHAEL CORRADINI

STEVE SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

KENT HOWARD

DEREK WIDMAYER

ALSO PRESENT:

DON ALGAMA, RES

DREW BARTO, NMSS

AMY CUBBAGE, NRR

RICHARD LEE, RES

SCOTT MOORE, Executive Director, ACRS

KIM WEBBER, RES

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P R O C E E D I N G S

9:30 a.m.

CHAIR BLEY: Good morning. This meeting will now come to order. It's a meeting of the Advisory Committee on Reactor Safeguards Subcommittee on Future Plant Designs.

I'm Dennis Bley, Chairman of the Future Plant Designs Subcommittee. ACRS members in attendance are Ron Ballinger, Charlie Brown, Vesna Dimitrijevic, Walt Kirchner, Jose March-Leuba, Dave Petti, Joy Rempe and Matt Sunseri will be joining us in about an hour. And our consultant Mike Corradini is in attendance for part of the meeting this morning.

Derek Widmayer of the ACRS staff is the designated federal official for this meeting. Kent Howard is the backup DFO for the meeting.

The purpose of today's meeting is to review the draft NUREG Document NRC-Non-Light Water Reactor Vision and Strategy, Volume 5, Radionuclide Characterization Criticality, Shielding and Transport in the Nuclear Fuel Cycle.

It's the final volume of the staff's documentation of their near-term implementation action plan for Strategy 2, computer codes.

The subcommittee will gather information, analyze the relevant issues and facts and formulate proposed positions and actions as appropriate. This matter will be brought to the February 2021 full committee meeting along with Volume 4 of the NUREG series for a possible letter report.

Previously on November 4 of 2019, we sent a letter report to the Chairman of the NRC from Volumes 1, 2 and 3 in an overview report. At the end of the today's subcommittee meeting, the members of the subcommittee and the staff will discuss plans for the February 2021 full committee meeting.

ACRS was established by statute and is governed by the Federal Advisory Committee Act, FACA. The committee can only speak through its published letter reports.

We can hold meetings to gather information and perform preparatory work that will support our deliberations at a full committee meeting. The rules for participation in ACRS meetings including today's were announced in the Federal Register on June 13 of 2019.

The ACRS Section of the U.S. NRC public website provides our charter, finalized agenda, letter reports and full transcripts of all full and subcommittee meetings, including the slides to be presented here.

The meeting notice and agenda for this meeting were posted there. And as stated in the Federal Register notice and in the public meeting notice posted to the website, members of the public who desire to provide written or oral comments to the subcommittee may do so and should contact the designated federal official five days prior to the meeting as practicable.

Today's meeting is open to public attendance, and we have received no written statements or requests to make oral statements.

We have also set aside 10 minutes in the agenda for spontaneous comments from members of the public attending or listening to our meetings. Due to the COVID pandemic, today's meeting is being held over Microsoft Teams for the ACRS and NRC staff attendees.

There is also a telephone bridge line allowing participation of the public over the phone.

A transcript of today's meeting is being kept. Therefore, we request that meeting participants on the bridge line identify themselves when they're asked to speak and to speak with sufficient clarify and volume so that they can be readily heard.

At this time I ask that attendees on Teams and on the bridge line keep their devices on mute to minimize disruptions and unmute only when speaking.

We will now proceed with the meeting. And I call on Kim Webber, Deputy Director of the Division of Systems Analysis in the Office of Research to begin. Kim?

MS. WEBBER: Yes. Good morning, everybody. I hope you all had a nice Thanksgiving. I know that I am still eating turkey. And I've been eating it since last Sunday, so I'm getting tired of eating leftovers. But anyway, hope you all had an enjoyable holiday and with that I'll get started on my presentation.

First, I want to thank you for taking the time to review our latest volume on code application activities. It's Volume 5, Radionuclide Characterization, Criticality, Shielding and Transport in a Nuclear Fuel Cycle.

My name is Kim Webber. I'm the Deputy Director of the Division of Systems Analysis in the Office of Nuclear Regulatory Research. And we will be asking for a letter on both Volumes 4 and 5.

Volume 4, you may recall, we presented to you, I think it was last month. And so I think we're also anticipating a full committee meeting sometime in the late winter time frame, maybe February or March.

Next slide, please. Okay. So with me today are Don Algama, he's the Senior Reactor Systems Engineer in the Office of Research, and Andrew Barto, a Senior Nuclear Engineer in the Office of Nuclear Material Safety and Safeguards.

They've been working very hard over the last several months to develop a strategy that we believe is the best approach to enable our readiness to support safety reviews of the front and back end of the fuel cycle.

Over the next few minutes, I'll provide an overview of the status of the non-light water reactor code development project and a short overview of Volume 5.

Then I'll turn the presentation over to Don and Drew, who are going to discuss the details of Volume 5, including the topics shown on this slide and in the agenda.

Could I have the next slide, please? RES's mission now more than ever is to enable the regulatory offices, like NRR, to be ready to perform licensing reviews and oversight responsibilities for advanced non-light water reactor technologies.

With that be ready attitude, we're doing research differently, embarking on more be ready strategies.

To improve mission value, we're working hard to deliver the tools, expertise and information in a cost effective and efficient manner so that licensing can be completed on time and within the allotted resources.

A key element of this strategy, as you know, is developing the codes and analytical tools. Direct code development activities and collaborations with many organizations you see on this slide were gaining knowledge and building staff expertise and analytical capabilities to support safety analysis for a wide range of advance reactor designs.

Next slide, please. To facilitate the Agency's readiness, the NRC's near-term implementation action plan was developed in 2017. The IAP is the vehicle to execute the NRC's vision to safely achieve effective and efficient non-light water reactor mission readiness.

As you know, the IAP includes six strategies and Strategy 2 focuses on computer codes and knowledge to perform regulatory reviews.

Next slide, please.

MS. REMPE: Kim?

MS. WEBBER: Yes?

MS. REMPE: This is Joy.

MS. WEBBER: Hi, Joy. Good morning.

MS. REMPE: Good morning. I had a question, and I couldn't decide whether to ask later or to ask you. But I think it pertains more than to just Volume 5 so I think I'm going to ask you.

In our biennial report last time we issued it, we recommended that RES review and update as needed the Agency's non-LWR implementation action plans to ensure that they emphasize the data that design developers have to obtain to validate codes for various new concepts.

And in the back of Volume 5, or I guess actually it's on Page 13, there are some statements that talk about the designs haven't provided enough detailed information on non-LWR fuel cycle implementations and so they realize that what they're doing may have to be updated.

But we observed the need for updates because when we started this non-LWR activity, there were very few details and the designs have evolved. And have you guys talked about when you think you're going to be updating some of these plans, how often they need to be updated? Or what's the trigger for trying to go back through and say what's still applicable and what's not applicable or what else needs to be added?

MS. WEBBER: Well, thank you for the question. So generally our strategy involves developing what we call reference plant models. And so those reference plant models are based on publicly available information of advance reactor designs that are very similar to the ones that, you know, we anticipate receiving.

So, for example, heat pipe reactors, we have a reference plant model for heat pipe reactors, sodium fast reactors, high temperature gas reactors, et cetera.

And those reference plant models are being developed not only in the context of the safety analysis work of Volume 1, but they're being developed in the context of Volume 3.

And the whole purpose for taking that approach is to minimize the amount of time that it would take to update the codes for design specific information.

And so the plan really is that these reports represent the global strategy and identify the gaps that exist and the verification validation needs, et cetera. But that really, when it comes to doing the design specific work, we're going to rely on our existing user need requests and RER research assistant request processes to, you know, do the more design specific licensing work.

So that activity will not be incorporated into any revision of these volumes. Does that help answer the question?

MS. REMPE: Yes. But so let me rephrase in a way to make sure I understand.

MS. WEBBER: Sure.

MS. REMPE: I was aware of the reference plant evaluations. And so you're going to use that to ensure that these volumes are sort of applicable. That you're not going to ever update these volumes because you will rely on what you learned from the reference plan evaluations and design specific activities to see if there are any gaps, and you'll deal with it elsewhere. But it sounds to me like you will not be updating these volumes. Is that a good conclusion from your response?

MS. WEBBER: Yes. I would characterize it slightly differently. So while these volumes represent what we know to be the gaps today and the verification/validation needs and the code development tasks, you know, they were developed at a point in time. And I would anticipate that unless there's a substantial change relative to the information that's contained in them that we will not need to update these volumes.

But like I said, if there is a substantial change, then one way to communicate our plans to reflect that substantial change would be to update whatever volume is needed.

MS. REMPE: Okay. So the reference plan evaluations may identify the need for a substantial change, et cetera, or some new design that you have to deal with may identify the need for a substantial change. But that would be the only reason that such a substantial change would occur.

MS. WEBBER: Yes. Like none of these volumes address fusion reactors, you know. And so there are things that are probably out there a little bit farther that when we started this work we did not envision like fusion technology.

And so, you know, if that becomes a reality then we'll have to start, you know, thinking a little bit more deliberately about how we address the gaps and the needs relative to, for example, fusion technology.

MS. REMPE: Okay. This helps. Thank you.

MS. WEBBER: You're welcome. Thank you.

CHAIR BLEY: Kim --

MS. WEBBER: Yes.

(Simultaneous speaking.)

CHAIR BLEY: -- just a little further there. First, I would like to thank you for this slide with the hot links to your updated volumes.

MS. WEBBER: Oh, good.

CHAIR BLEY: And I don't know if anybody has done that before so I appreciate it.

MS. WEBBER: Well, I've got to thank my staff for doing that.

CHAIR BLEY: Well, the introduction, Volume 1, Volume 2, Volume 3, were issued in these versions in January. I haven't been through those yet. But are they updates of the ones we reviewed a year ago?

MS. WEBBER: Well, so you may recall that you -- I'm getting a weird echo. You may recall that we issued the introduction, Volume 1, 2 and 3 and had a meeting with you last November of 2019, I believe. And then we updated these volumes to reflect comments and feedback that we received through the various meetings and also as a result of that letter.

And so the versions that you see for the introduction, Volume 1, 2 and 3 are the final set that reflect modifications, the feedback that we received from you. Now Volume 4, we had the subcommittee meeting in, I think it was October.

CHAIR BLEY: Late September, but go ahead.

MS. WEBBER: Yes, late September. So this one is still a draft. And the staff, I know that they recently looked at the transcript. And so they're trying to update that volume, you know, as we speak. And then if we go into the full committee meeting, they'll take whatever feedback from that.

And Volume 4 and 5 together, we will finalize in a version that's, you know, sort of the official Version 0 or Version 1.

So, you know, if you could see these pictures on Slide 5 for the different volumes, you would note that there's a date in there of, I think it's January.

CHAIR BLEY: That's right.

MS. WEBBER: Yes. It's January. And so that represents sort of the final Version 1 of these documents, at least at this point.

CHAIR BLEY: So Volume 4, well, I guess looking through the slides that the gentlemen are going to provide next, it looks like you made some presentations on kind of changes since Volume 5 was published.

Do you expect you will revise to any extent Volumes 4 and 5 before our February meeting?

MS. WEBBER: Well, we probably will make some revisions. And, you know, if you're interested in seeing, like, a red line strike out version of the two volumes before the full committee meeting, we would be happy to provide that if that would --

CHAIR BLEY: Thanks. That would be very helpful. We would appreciate that.

MS. WEBBER: Okay. Yes. We could do that.

CHAIR BLEY: Okay. One last question in this area, and we won't talk about it at the end of the meeting. The introduction was pretty thin when we saw it the last time, and we noticed some inconsistencies in approach in Volumes 1, 2 and 3.

Were those addressed and should we -- at the February meeting, would it be worth 15 minutes to half an hour to bring us up to date on what you changed in introduction, 1, 2 and 3?

MS. WEBBER: Yes. We could do that. You know, maybe we need to talk offline about the specific interests that you have because I'm not clear on the specific interests relative to doing that.

CHAIR BLEY: Okay. Well I'll have Derek work with you and set up something to talk about that because that might affect how we decide to write the letter come February. Sorry for all the interruptions. Go ahead.

MR. PETTI: I had a question. This is Dave. Since we're talking about the big picture here.

MS. WEBBER: Mm-hmm.

MR. PETTI: I think it's hard to write Volume 5 so I don't want this to come across as critical.

MS. WEBBER: Mm-hmm.

MR. PETTI: But I'm trying to understand the backdrop here. You guys are envisioning, for instance, fuel fabrication facilities and doing criticality analysis of new fuel fabrication facilities for advance reactors, which have different fuels and LWRs. That seems to be something well downstream in the future --

MS. WEBBER: Mm-hmm.

MR. PETTI: -- compared to said Volumes 1, 2 and 3 where, you know, the first reactor you're going to do something with. The document is silent on the fact that the first cause for these reactors are probably going to come from down blended HEU.

MS. WEBBER: Mm-hmm.

MR. PETTI: It would have been made by DOE or by commercial vendors that have a license to handle HEU and HALEU. And so it kind of just, it threw me. It would seem to me that a footnote or a paragraph that recognizes where we are today relative to sort of where you are envisioning it, you know, in a full, you know, commercial setting --

MS. WEBBER: Sure.

MEMBER PETTI: -- where you've actually got more than one would probably help because, you know --

MS. WEBBER: Okay.

MEMBER PETTI: -- I mean, I didn't hear anybody is much more focused on, you know, I need a -- I need HALEU now and that's a whole different conversation. And then you read this, and it just struck that you guys know this but the document doesn't talk about that. And it makes it seem a little, like, you know, out in left field.

MS. WEBBER: I think that's a good comment. I think that's a good comment. And I appreciate you for bringing that up. And that's probably something that we can address in the revision to the report.

MR. PETTI: Okay. Yes. Because there's a number of things like that where just footnotes probably would help to just clarify some things -- I'll go through others as the slides come along so.

MS. WEBBER: Okay.

MEMBER PETTI: Thanks.

CHAIR BLEY: Yes, this is Dennis. One last time, Kim. What Dave brought up resonated with something that I've been thinking about. And this is no surprise because these are delving into, in some cases, into new areas.

It seems like the 10 reports you're going to tell us about that are coming out of this plan -- this is substantially different than especially Volumes 1, 2 and 3.

MS. WEBBER: Yes.

CHAIR BLEY: This is a plan, and those 10 reports are going to eventually get us to the kind of evaluation you did for the other codes in 1, 2 and 3. Is that correct?

MS. WEBBER: Yes. Yes, conceptually, I think that's what's going to happen.

CHAIR BLEY: Yes.

MS. WEBBER: But I think due to the complexity of the fuel cycles for each of the different designs and all the subtleties and nuances, you know, I think the staff has done a really good job of at least identifying, you know, the strategy in the Volume 5 report.

And so, you know, once the strategy has been identified, then I think they can focus more specifically on a particular fuel cycle of interest or a different step. And I'm kind of jumping ahead into, you know, Don and Drew's presentation. But I think it's at least a good start at a strategy to figure out how best to do this.

And as, you know, you probably are aware, a lot of the information on the fuel cycles is still, you know, to be determined. And so we're really kind of leaning forward to do the best that we can to figure out what our information needs are and our, you know, model development needs are.

And so, you know, this particular volume is likely, you know, to evolve over time or the strategies. And, you know, Don and Drew will talk about it. But, you know, we're going to have to prioritize based on, you know, what we see as the most important steps.

And we're already getting indicators, you know, that people want to ship fuel for these designs. I just heard the other day about someone being interested in designing a package to ship fresh TRISO fuel. You know, so the activities are already being thought about.

You know, there's regulatory efforts that are underway. And I'm pretty sure Drew can answer some of the more detailed questions you might have during his presentation. But if it's okay with you, you know, let me just finish up my next few slides and then we'll get into the details of Volume 5.

CHAIR BLEY: Okay. Go ahead.

MS. WEBBER: Okay. One last comment, I think, Dennis, you had the question about, you know, what the full committee meeting and the letter will focus on. So we do have a letter on the introduction, Volume 1, 2 and 3 and what we're seeking more specifically is a letter on Volume 4 and 5.

So originally the thought was not to necessarily go back and do a reassessment of the intro, Volume 1, 2 and 3, but it was to really focus on Volume 4 and 5. So that was at least my initial thought, but we can talk about that.

So I think I've touched on, you know, the information relative to this slide. I guess the only other thing that I wanted to point out for those in the audience who may not have as much familiarity is that, you know, Volumes 1 through 3 focus on the systems analysis, fuel performance, neutronic source term, severe accident progression and accident consequence codes.

And then Volume 4 describes code development plans for our suite of codes used to evaluate the siting criteria, control room inhabitability and other safety evaluations during licensing. And then we talked about sort of the focus for Volume 5 so we'll go to the next slide.

So, you know, if you'd like to follow the status of our code development activities, you can go to the advance reactor on our see public web page, which is shown at the top left corner of this slide.

And then if you scroll down to the page and then click on the summary of integrated schedule and regulatory activities image, which is shown in the bottom right-hand of this slide, then you'll see the status of the major milestones for the near-term code development tasks.

And a large portion of what we're doing for Volumes 1 and 3 are these reference plant models and building them out. And I think, you know, the plans are to have much of that reference plant model work done this year in 2021.

So can we go on to the last slide in my presentation? So Volume 5 describes the staff's plans to evaluate the ability of scale in MELCOR to support safety analysis and licensing for front end and back end of the fuel cycle.

By considering the fuel cycles for many non-light water reactor designs, the staff developed an approach that involves evaluating information gaps and identifying methods that can be used to address the gaps.

Using the light water reactor fuel cycle as a reference point, the staff plans to develop a series of individual reports, which we had been talking about, and publicly available input decks that characterize the co-development needs for all aspects of fuel fabrication, transportation and storage as we know them.

And, you know, due to the dynamic nature of not only the advance reactor industry in terms of designing their reactors, but there's also an extremely dynamic fuel cycle process for each one of those plant designs as well.

And so now unless there are any questions, I'll turn the presentation over to Don.

MR. CORRADINI: So, Kim, this is Michael Corradini.

MS. WEBBER: Yes. Hi, Mike.

MR. CORRADINI: Hi, how are you?

MS. WEBBER: Good.

MR. CORRADINI: I hope you had a nice holiday.

MS. WEBBER: Yes, it was great.

MR. CORRADINI: My big picture conclusion from reading the volume and looking at your slides is that the basis will be no core max and the current tool scale.

MS. WEBBER: Yes.

MR. CORRADINI: And there will be slight modifications as needed, but the overall structure is already in place.

MS. WEBBER: Yes.

MR. ALGAMA: Yes.

MR. CORRADINI: Okay.

MS. WEBBER: And Don can talk -- I think Don and/or Drew may talk more about that, Mike.

MR. CORRADINI: All right. I'm sure. I just wanted to make kind of the 40,000 foot conclusion is clear because I think that's personally the way to go. Some of the suggestions Dave made might be appropriate given the fact where the initial fuel loadings will come from. But, okay. Thank you very much.

MS. WEBBER: Yes. So just to expand on that a little bit, as you know in Volume 1, we're using new codes, Department of Energy funded codes. But like Volume 3, we're going to use our own, you know, well-known codes and filling gaps wherever those gaps may exist. All right.

MR. CORRADINI: Thank you.

MS. WEBBER: You're welcome. All right. I'm going to turn it over to Don now.

MR. ALGAMA: Thank you. Hopefully I can change. Oh, there we go. Can everyone see the slides?

MS. WEBBER: Yes.

MR. ALGAMA: Thank you. Howdy. My name is Donald Algama and I'm with Drew Barto. Today we're here to discuss Volume 5 as Kim as already provided.

It is important to note that this is a plan. And as we learn more during the process, especially implementation and gathering information from the DOE and vendors, we will update the implementation part of the plan as we move forward.

Sorry. It starts changing. Oh, there we go. Okay. This is an acknowledgment to all the great help we received from both the program officers from NMSS, NRR and research and also David Luxat from Sandia and Will Wieselquist from Oak Ridge. So thanks for all the help in doing this.

You've already seen this part so I'll skip over this. This is just a summary of the IAPs to date. And with this, we start.

The goal is to apply and understand the performance of existing NRC tools to support fuel cycle evaluations. And the intention is that we will gain experience in all fuel cycles and at the same time demonstrate computer code readiness.

As a plan, it is intended to be updated as we learn more from DOE and the industry for both the designs and what they may be expecting from their normal fuel cycle approach.

This plan will take on a delta approach using the existing LWR fuel cycle as a reference. Basically, an incremental approach comparing the candidate and non-LWR design against existing fuel cycle capabilities.

As we are taking an LWR approach, in

practice this means core knitting with internal partners when scenarios demonstrate the need such as those in Volume 3 and Volume 4 and our NMSS teams concerned about release, dose, materials, et cetera.

Volume 3, the impacts using this work will be made public. This plan leverages LWR experience to the extent possible. Thus, the following few slides will provide an idea of how these codes are used in the existing framework and existing staff experience.

The red box highlights areas in the LWF fuel cycle as a potential use in this work. The following two slides will provide further examples.

This slide provides an overview of the transportation of storage space as of today. The slides start from fundamental nuclear data, processing the application to scale and then possible follow-on work.

In this area, scale is currently being to the context of criticality and shielding for spent fuel package designs and for spent fuel dry storage systems, shield analysis to support radioactive material process and package designs and for dry storage systems including the waste consolidation storage and Holtec HI-STORE Consolidated Interim Storage Facility applications.

It's also been used in transport, criticality analysis for packages of UA6, U02 powder and pellets, commercial and research, fresh and spent fuel assemblies, et cetera.

MEMBER MARCH-LEUBA: Don?

MR. ALGAMA: Yes?

MEMBER MARCH-LEUBA: This is Jose.

MR. ALGAMA: Hi, Jose.

MEMBER MARCH-LEUBA: Yes. Have you thought about the uncertainty of core second generation? For a long time core second generation was an art. It has now become more of a science but that's because of all the experience we have with configuration with fuel rods and light water. And we have resolved all the problems.

But when you are going to these unusual configurations like a molten core or even a little bit of the pebble reactors. So have you given consideration to uncertainty of cross-sections?

MR. ALGAMA: Yes. That will be considered in the implementation phase in part of the 10 reports.

MEMBER MARCH-LEUBA: And is there going to be sufficient data to benchmark criticality?

MR. ALGAMA: Yes and no.

MEMBER MARCH-LEUBA: Okay.

MR. ALGAMA: So as of right now for the HALEU space, we are developing approaches to mitigate the lack of benchmark data or appropriate benchmark data, but we'll be evaluating those as we go through the implementation phase.

Will Wieselquist can answer more if he can, but we'll be evaluating it. But we haven't really got there yet.

MEMBER MARCH-LEUBA: Okay, yes. You need to give it some thought because if there is need for experimental data for a particularly unusual configuration for which we don't have any experience that would be really bad because we --

(Simultaneous speaking.)

MR. ALGAMA: Yes. Understood.

MR. BARTO: So this is Drew Barto. I don't think Will is on the line. But I can try to answer for him. You know, that is a very good point. And that's a big part of what we'll be looking at in terms of gaps. You know, really moving forward we've used these tools for a number of years, you know, mostly for LWR type of analyses.

But we really have been able to evaluate some of the materials and configurations that are going to be used in the advance reactor fuel cycle.

So we've been able to -- as far as the codes themselves, they have the capability of modeling these things, like, you're right. None of that means anything if you can't validate it.

And so that's a very important part of what we'll be looking at. You know, what experiments are available? You know, to what extent can you use experiments?

You might now think it looks like your system, but neutronically they are similar so there's lots of use of say, sensitivity and uncertainty analyses, methodologies to compare critical systems.

So, you're right, that is a very important part of this.

MR. PETTI: So are you guys hooked into the criticality benchmark, IAEA activity where they have housed tremendous amounts of data on criticality and other similar experiments across the reactor spectrum so there's been tons of gas reactor stuff that I'm aware of, fast reactor stuff that you guys could, you know, check tools against?

MR. ALGAMA: I will look into it. I'm not aware of this off the top of my head.

MR. PETTI: It's a huge -- I mean, it was a, I don't know, three or four person effort probably in the U.S. alone, and it's international in its scope so.

MR. ALGAMA: Is it different than to the OECD benchmark?

MR. PETTI: No, no, no. I'm sorry. OECD is what I meant, not IE --

MR. ALGAMA: Oh, yes, we're aware of that, yes.

MR. PETTI: Yes, yes. There's a lot in there so.

MR. ALGAMA: Yes, sir.

MR. PETTI: Yes.

MR. ALGAMA: And we used that in part of our valid suite, too, for validating scale or setting scale's performance.

MEMBER REMPE: Don?

MR. ALGAMA: Yes, ma'am.

MEMBER REMPE: This is Joy. I had a question or comment. I was looking through the report, and I'm not sure how you would address it, but I think a paragraph is worthwhile to add to the report about these reactors that are supposed to be fabricated in a different facility and the core loaded and then transported and installed at a site and then removed from the site and taken somewhere for whatever they do to unload the fuel.

Because I assume it would be covered in this Volume 5 activity, but it's not really discussed or I missed it if it was discussed in the report and what you plan to do on it. And I'm not sure what you would do, but perhaps it ought to be acknowledged that this something that may have to be considered.

MR. ALGAMA: Understood.

MEMBER REMPE: But what are your thoughts about what you would do with something like that?

MR. ALGAMA: Going through the fuel cycle, I think the intention was the -- I think the tables -- we provided the flowchart of analysis within.

MEMBER REMPE: Right. And I --

MR. ALGAMA: That would be where we discussed those kinds of activities. So we start --

MEMBER REMPE: So I looked for that, and I did not -- again the way the sodium fast reactor because one of the ones they're talking about, I did not see it there or in any of the others where it just called out and said we need to think about this type of structure where you would actually have -- they talk about loading the core at the site. They don't talk about loading it offsite and transporting it to the site, right? I did not see that in one of those flow diagrams.

MR. ALGAMA: I understand. So that was the difference between the HPR and the SFR cores, where the SFRs had a, like, a regular LWR approach where their centers would be manufactured and then shipped out to the site for loading. And then the HPR where we anticipate that the whole reactor core will be fabricated in the fabrication site and then shipped out.

We did try to put some text in the report about the two different approaches, but we can add more to be --

(Simultaneous speaking.)

MEMBER REMPE: Maybe I missed it. But, again, I think that that is something that may -- I mean, do our existing tools cover something like that?

MR. ALGAMA: Existing tools cover -- I'm not sure. Forgive me. Could your rephrase the question?

MEMBER REMPE: Well, do we think about transporting -- I mean, can you use scale or something to deal with a criticality event when you have a loaded core being transported somewhere and installed on the site?

MR. ALGAMA: Yes.

MEMBER REMPE: I mean, because we have the tools and capabilities for doing that we just haven't ever applied them for such a situation?

MR. ALGAMA: Correct. Yes, we can apply the tools. But like Jose was saying, we have to be careful on what the results mean, developing an appropriate validation basis and uncertainty analysis to go with it. But yes, the short answer is yes.

MEMBER REMPE: Okay. So I just think that we need to discuss that a bit more in the report to acknowledge that we're thinking about it, but, you know, it's something that will be addressed or something. You know, I guess I did not see that enough when I was looking in the text but maybe I missed it.

MR. ALGAMA No. We can add more. Thank you.

MR. PETTI: Okay. This is a case again the assumption on the heat pipe reactor, I understand where it came from. But there's another heat pack reactor potentially, at least a microreactor that it's different enough that it may cause you to rethink a little bit how the different pieces fit together.

And that's what I kept struggling with is in general you have to make a number of assumptions, right, to kind of weigh this out. What if you're assumptions are wrong and how would that impact, you know, the approach? It would just seem like it would be worth a little bit of thinking about that. I don't think it will change the fact that the tools, you know, can do the job. It's just, you know, your view of the future may not be exactly what the future is.

MR. ALGAMA: Understood.

MR. PETTI: Right. So, I mean, it might be worth just a paragraph or even a footnote of that that, you know, even though this is what we've said, we think, you know, more broadly that the tools can handle, you know, some sort of evolution away from these assumptions so.

MR. ALGAMA: Yes, sir.

MR. BARTO: Hey, this is Drew. And I'll just add to that. I think you're right, it could benefit from a little more discussion. And I think as far as neutronics tools for criticality and shielding that it's not going to be that much of a challenge to model, you know, whatever comes forward in terms of heat pipe reactors or other transportable reactors.

The challenge with those is really going to be in the structural and thermal analysis showing that they can survive the 10 CFR Part 71 transportation accidents, which I'm sure you're aware are much more challenging for a stationary system.

So it's going to be showing that the system can withstand those accidents and then translating that into a configuration that your re-tracks tools can model. And is that configuration appropriate? And that's really going to boil down to, I think, the nuts and bolts of an actual technical review. But it should not be a challenge for the scale or the other tools to model such configurations.

MR. PETTI: Right. Thanks.

MS. WEBBER: But the one thing I want to note. I do agree that it's worth adding, you know, some information about that configuration, you know, with the fuel loaded into the reactor and then the whole reactor with the fuel shipped to wherever it's going. So I think that's something that we can do.

MR. ALGAMA: It's more of a story of what we anticipate and how we would accommodate changes.

MS. WEBBER: Well, the nuances of that particular type of reactor design, microreactors.

MR. ALGAMA: Okay. I'm going to move to the next slide. Is that okay? I take that as a yes.

Just so that I capture the basis of Volume 3 approach from our analysis as you've seen before.

As before, fundamental data is processed and applied by SCALE and passed as input a severe -- as input of a severe accident and source term code MELCOR and offside analysis code MACCS.

The following slides are some examples of starting fuel cycle experience applying the scale of MELCORs to non-reactor facilities in transport and storage areas.

The codes have been applied in the L3 PRA project. And here at 2161 is the spent fuel core study at NUREG 7108 and 7109, which is the developing estimates on isotopic depletion bias and uncertainty and criticality uncertainty.

This is a recent application of scale in MELCOR to a non-power facility. This analysis looks at a range of scenarios at the Barnwell Nuclear Fuel Plant and the effectiveness of various plans of defense within the reprocessing facility.

Five of the classes of accidents in the FSA were evaluated with the scale MELCOR package. And we captured material degradation, building leakage, aerosol physics for deposition, agglomeration, et cetera. And we also looked at leak path factor considerations, impacts of filters, ventilation systems, instructs as a result of fires.

MEMBER PETTI: I had a question back on the burner credit. You know, some of these burnups significantly beyond what we think of in the light water reactor context.

MR. ALGAMA: Yes, sir.

MEMBER PETTI: Do you guys have any idea how good the SCALE code suite will do? Because, you know, you're going to be fissioning a lot more plutonium as you get those really high burnups and the uncertainties of the fissioning of the higher actinides?

MR. ALGAMA: Yes. So we're actually pursuing research as part of ATF/HBU to see if we can develop methodology that would extend or depletion and uncertainty analysis along with that.

We would eventually need validation data to see just how good we are, but we have an approach in mind.

MEMBER PETTI: So there is data, very recent data, for gas reactors. And I think there's probably similar data for a fast reactor fuel as well. So it's just a matter of getting access to it.

MR. ALGAMA: Yes, sir. You wouldn't by chance have the reference for that do you?

MEMBER PETTI: Well, the HER program has

published the burnup comparisons with actual destructive burnup and measurements of season fission product ratios correlated to burnup. So that's out there in the public literature. And the fast reactor stuff is a little bit older because we haven't had a fast reactor in the U.S. But I'm sure there's data from EBI, too --

MR. ALGAMA: Yes.

MEMBER PETTI: -- that would be useful so.

MR. ALGAMA: I see.

MEMBER PETTI: Yes.

MR. ALGAMA: Thank you. Let's skip over this one. So this slide is a copy of Table 1-1. The intention is to provide a high level of understanding of what differentiates non-LWRS and LWRS right now.

Some notable features are that the designs are based on uranium and share front end UA6 enrichment needs that are common and some fabrication needs that are common.

Fuel forms range from oxides and metals to uranium dissolved in molten salts. The neutron spectrum can be firm all the way to fast. Burnups, as you mentioned, Dr. Petti, can be very large compared to LWRs and numbers that potentially include onsite fuel processing. So all these things will have to be evaluated.

As mentioned the objectives in this plan and its resulting reports ultimately demonstrate computer code readiness. To achieve this, we will have to look at developing scenarios and identify potential hazards to assess the codes against.

We intend to look at available NRC, DOE and design information as they come up to help understand the potential on non-LWR fuel cycle. And thus this plan will evolve as we implement as well as historical information.

MEMBER REMPE: Don?

MR. ALGAMA: Yes.

MEMBER REMPE: I didn't meant to interrupt you. Go ahead and finish. But I have a question when you finish this slide.

MR. ALGAMA: Yes, ma'am. Hazard evaluation, there are documents that can be used to develop scenarios to test core performance in criticality safety, our inventory characterization indicate heat estimation, radiation shielding and RN, radionuclide and other hazard evaluations.

Further analysis needs -- consequence analysis areas will be raised to the appropriate team at NSNRI within Volume 3 and 4 as they occur.

We will use NUREG 6410 to drive our scenario selection for fuel cycle facilities. And in particular, it includes a process hazard analysis approach, which is a technique to identify and understand scenarios that merit further analysis.

This handbook, 6410, covers criticality events, release of materials, in-facility transport depletion processes, leak path factors. And Table 2 of that provides a range of scenarios that could be considered for existing facilities.

In 1520, which compliments 6410, the purpose of the SRP is to ensure quality and uniformity of reviews, which also provides further insights on how we should assess our codes.

In 2015, the move from facilities to transport. And this NUREG focuses on COC for dry storage systems and ISFSIs and monitored retrievable storage installations.

In 2016, we moved towards transportation, which covers fueling criticality, et cetera, and provides a -- Table 1-2 of this report provides an example of scenarios to demonstrate some criticality. And Attachment 2A provides staff expectations of computer codes.

Moving along, there are complementary DOE documents that we could leverage. One as an example that may be useful to develop hazards is listed. The other documents such as DOE Standard 1027 has an evaluation techniques, DOE Standard 2007, which covers SERs for non-power facilities, et cetera. These will be all reviewed in the implementation phase.

So an example scenario may be an accident at a fuel fabrication facility. An accident occurs where -- I hypothesize, where the UA6 cylinder is damaged while it is in the process of being evacuated. Staff may be interested in investigating possible UA6 release, chemical reactions from the damaged canister and into the facility environment.

Joy, I'm going to move to the next slide so you had a question?

MEMBER REMPE: Yes. First of all, earlier I meant to tell you I really like Slide 5 and Slide 10. I thought those were nice slide summaries of how codes were used for those regulatory activities and where there were gaps.

But when I was looking in your report and thinking about how you're going to develop scenarios, I think it might behoove NRC -- I'm not as familiar with this DOE handbook. But it might behoove NRC staff to think about a more in-depth review of prior experience that's more recent.

The Tokaimura accident happened in 1999 but 6410 was a lot older as I recall. You mentioned you've got a lot of experience, the Agency does, with non-LWRs and you go back and mention this being an L report. But it's a very high level summary report that rarely go into depth of things that have happened with gas reactors like Fort St. Vrain as well as Fermi.

And there are a lot of times where lack of administrative controls have led to fuel melting and severe situations like what happened at Tokaimura. And I am wondering if maybe some more in-depth review is needed unless there's something in this DOE Handbook that will give you some really good ideas about scenario selection. What are your thoughts about that?

MR. ALGAMA: No, no. I one hundred percent agree. That was the intention also was to look at historical data to guide us in what would be -- hazards of interest to apply our codes and see how they perform.

MEMBER REMPE: Yes. Because I do think there's some very good lessons in history. But I just haven't seen enough discussion of that. And so it might behoove you to go a little more in-depth. Bad things have happened when people do things without enough review and don't have enough administrative controls. And I'll stop there.

MR. ALGAMA: Yes, ma'am.

MS. WEBBER: So, Joy, just to make sure I understand your comment. So are you suggesting that in the report that there's maybe a little bit more about scenarios that need to be evaluated in the context of the scope of the report?

MEMBER REMPE: I think the report is fine. But I think maybe research might want to think about -- again it depends on how the future plays out. But if we're going to try and do this for non-LWRs, I think a more detailed review of what's happened in the past would behoove us.

MR. ALGAMA: Could I just state one -- I'm sorry.

MEMBER REMPE: Yes. And then, again, when you don't have the details of these new facilities because they're just conceptual ideas, it's hard to do that. But I think those things -- you know, again, I recently was involved in a project where we looked more in-depth of what happened at Fermi 1 and Fort St. Vrain with its startup.

It's just when there's not enough administrative controls, there's not enough review, things have happened. And Tokaimura is an example where, again, people applied something, a process they had used for a lot of times to something a bit different. And people didn't, you know, have enough oversight and review of the situation before things occurred.

And so, again, I was interested in your report. And you mentioned, oh, you've got this Brookhaven report. And there's barely a paragraph about each reactor.

And I think somebody needs -- I'm sure there's people around, and there's a lot of history around. And I just think it might be a good thing for research to do if this whole non-LWR thing comes to fruition.

MR. ALGAMA: Would that be something we would consider an implementation phase? That was the idea at least.

MEMBER REMPE: Yes. I think, I mean, you might acknowledge that clearly a more in-depth review would be performed because of situations in the past. But I just think that a more detailed review would be good.

And how you want to address that, again, I wouldn't go spend money on it today unless we know for sure somebody is going to do this, but I think a more detailed is needed at some part. And it's up to you guys how you take that. It's just one member's comment if you want to try and do something that way.

MR. ALGAMA: Understood.

MS. WEBBER: To me it sounds like really a, you know, broader operating experience review of all the technologies.

MEMBER REMPE: Yes.

MS. WEBBER: Okay. Thanks. I'm not sure it's really in the scope of this report. But where it's relevant, you know, we could, you know, add some additional text.

MR. ALGAMA: So once we are done with scenario selection, we move on to the scope of the analysis. With areas such as mining, milling, long-term storage and disposal consequences, radiation protection, chemical toxicity would be counted elsewhere.

CHAIR BLEY: I'm sorry. But my brain just caught up with --

MR. ALGAMA: Yes, sir. Do you want me to go back a slide?

CHAIR BLEY: No. This is for Kim and our past discussion. If we're looking at scenarios and the ability to identify them is crucial and if we don't look carefully at the history when missing a source of information to make that a more complete assessment, I don't see why it doesn't fit here, Kim.

MS. WEBBER: Yes, I guess. So in the context of the front end and the back end of the fuel cycle, you know, I think, you know, there's obvious relevance to this scope.

But I think what Joy may have been advocating, and correct me if I'm wrong, is something more broad about, you know, she mentioned admin controls and startup of the reactor. And so there's broader operating experience related to the operations of these reactors.

And so I think that the, you know, really what's relevant to the fuel cycle are the operating experience relative to the front end and back end of the fuel cycle. I think that's what I meant.

CHAIR BLEY: Okay.

MS. WEBBER: But thanks for the comment. I appreciate that.

MR. ALGAMA: All right. As with Volume 3, we expect to reasonably apply comprehensive methodological approach from scenario definition, identification of safety related items, identification of dominant phenomena to support that through the V&V and documentation.

We also intend on using the designs developed in Volume 3 to support fuel cycle analysis in Volume 5.

Continuing an example, it continues from the previously mentioned. Staff may want to know how the UA6 can be transferred in the damaged canister, how much HF is produced and where is the uranium deposited within the facility, specifically the HVAC to understand criticality implications, deposit materials, et cetera. We would deploy a combination of SCALE and MELCOR to try and evaluate that scenario.

Here, we move on to the 10 anticipated reports. Obviously, this would all be contingent on what we learned. We can adapt. We are flexible. As we learn more from the DOE and its partners, we can change how we prioritize the work in both 1, 3 and 5.

The term reports are broken down into five reports looking at non-LWR, specific fuel cycles and five reports that cover common fuel cycle activities.

The reason for this is to take advantage of commonalities. If you look at the HTGR and FHR fuel cycles, we can see that Reports 3, 7 and 10 are common. So once developed for one, it will be applicable to the FHR, for example.

MEMBER PETTI: So let me just -- if you go back. This is a common flaw throughout the whole report, a nomenclature problem, on Number 7 here, TRISO fuel kernel. The kernel as a nomenclature is the fissile part of the particle. But I'm sure you would read about the particle fabrication as well.

MR. ALGAMA: Yes.

MEMBER PETTI: So do you think you want to say kernel/particle or kernel and particle fabrication and just go through the whole report. And most of the time I think you mean particle. But there are a couple of times where I think you meant both, the fissile kernel and then the coated particle, just to use nomenclature that's more traditional.

MR. ALGAMA: Yes, sir.

MEMBER PETTI: Similarly, this is one of the assumptions that struck me was that you assumed that the fuel element here, you have it as a pebble, would be a different facility from where the particles are made. That has never, ever happened in the world. All of the Germans, the Chinese, the Japanese, the Americans all -- it's all in one facility.

There can be different material balance areas for sure to deal with accountability and the like, but they would not probably be large scale shipment of coated particles from one facility to another because they are actually fairly fragile in that state. And so it's always done in one facility.

MR. ALGAMA: Yes, sir.

MEMBER PETTI: So I would clean that up just so, you know, people wouldn't say, oh, they don't really know what's going on.

MEMBER KIRCHNER: Yes. Dave, this is Walt. I agree, yes. The nomenclature on seven should be more inclusive. And, yes, 10 as a standalone, then it begs the question what about compacts, which is the alternate means of taking the particle fuel and putting it into a serviceable form that can be loaded into a reactor.

MR. ALGAMA: Right.

MEMBER KIRCHNER: So, yes, I think these could be combined.

MEMBER PETTI: And then, you know, Kim is talking about shipping TRISO fuel. And it's compact. And that's a project that's underway right now. And so this is a case where you guys are trying to see the future, and, you know, it doesn't align with where we are today. So you could just say compact or pebble fabrication and --

(Simultaneous speaking.)

MEMBER KIRCHNER: Yes. I would combine them. When I bought fuel from GA, it was shipped to us in the form of compact. So it wasn't loose pebble particles.

MEMBER PETTI: Particles, right, right, so.

MR. ALGAMA: We didn't actually consider transport of TRISOs to a pebble facility. Will is on the line right now maybe he can add to this. But we did try to make a differentiation between pebble and fuel compact scenarios for the fuel cycle. Will, can you chime in a little bit? But we can make updates to the report to make it clear.

MEMBER PETTI: Yes. It would be interesting to know why you thought there was a difference, at least at the level that you guys are at --

MR. ALGAMA: Mm-hmm.

MEMBER PETTI: -- they look really similar. If you would have recycled the fuel in type of a cover uranium, things can get a little bit different. But they go through all the same steps.

It's just the geometry is instead of pressing a cylinder you're pressing a sphere.

MR. ALGAMA: Okay. And we referenced compacts, but we didn't look into it because at the time of this report we didn't have a driver for it. But that's something we can look at again.

MEMBER PETTI: Right. And now this one microreactor project the basis is TRISO and compacts.

MR. ALGAMA: Yes.

MEMBER PETTI: And then again, that's a thermal system. That's another thing that when you mentioned heat pipe reactor, you basically locked yourself into fast, a fast system, but they are thermal systems as well.

MR. ALGAMA: Understood.

MS. WEBBER: Thanks, Dave and Walt. I appreciate those insights.

MR. ALGAMA: So this leg, we begin our strategy. As mentioned, the LWR fuel cycle we use as a reference to understand the anticipated non-LWR fuel cycle. To make the task more tractable, we broke them down into six major steps and several stump steps.

These are labeled with the first step of the stage and a number for the substep.

So for fabrication we can break down the two steps, identify the F1 and F2. This work will not right now look at scenarios of interest in the T3 and S1 steps due to lack of understanding of where the DOE industry plans to go. That's probably way, way too far in the future for us. We will revise as we learn more.

The FHR class, the fuel cycle analysis, will be driven by the Berkeley Mark-1 FHR design as we had in Volume 3. The basic design uses TRISO particles up to 20 weight percent.

This directed design loads pebble from the bottom and are removed from the top. There are hundreds of thousands of pebbles that are expected to be used with thousands of TRISO particles each.

Rather than helium they will use a molten salt like FLiBe as the coolant. But the fuel cycle analysis stage, I expect it to be identical for what do for HTGRs but with some additional features such as moats for fission particle inventory migration within the coolant and then compared to HTGRs and tritium generation, transport and retention phenomena in both the FLiBe and the graphite.

Steps E1 and E2 will be completed in earlier reports as we described for commonalities. In E1, we will look at fresh fuel, how they will be staged and the expectation is looking at criticality type accidents here from fuel handling operations.

Step E2 is covered in Volume 3 where interactive data such as anticipated discharge relapse will be generated. This work may also consider radionuclide hazards during different fuel cycle operations and hazards with respect to fuel handling as I mentioned earlier.

In Step U3, it is not expected because we don't expect central fuel shuffle operations.

In Step 4, we expect onsite storage of spent fuel pebbles will be reviewed with respect to criticality, fuel and decay heat and other accidents.

For the HPR fuel cycle, it will be driven by a modified version of INL Design A, which comes from Volume 3. The basic design is the SFR and HPR are essentially the same in the front end of the fuel cycle, with the exception of how the fuel is actually manufactured.

Traditional SFRs have assemblies while HPRs are expected to be manufactured as an entire core but a bit smaller than an SFR core.

The fuel will be modified to be metallic. The INL design and discharge burnups increase around 10 gigawatt day MTU (phonetic).

The design is hexagonal with a sodium bond that thermally connects -- with a sodium bond to connect the fuel and the coolant.

In Steps E1 through F1, it will be done earlier. The work will start at the F2 stage, fabrication of the HPR core to reach transport to the utilization stage.

The F2 stage included the step due to the unique processes we anticipate when you're looking at developing a whole new core to transport.

The new stage of the core, the fresh core will be reviewed with respect to criticality concerns, staging areas, et cetera.

Stage U2 will make use of developments in Volume 3 and again also vary and are adapted for use in metallic uranium.

In the U4 stage, we will look at the full range of criticality shielding decay heat and hazard analysis.

The SFR fuel cycle reference reactor is under consideration still. Two possibilities stand out as the MET-1,000 benchmark design or the VTR. More information will be reviewed as we go into the implementation phase for this phase of the report.

Basic information is that this design can come with a wide range of fuel fonts from oxides, carbides, nitrides and metals. The metallic form will likely be a driver for this work. Enrichments up to 20% can be expected.

As before, Steps E1 through T2 will be covered in other reports. At U1 stage, we will look at criticality concerns mainly we anticipate for the fresh fuel assemblies. At the U2 stage, we will leverage the work that will be performed under Volume 3.

Unlike the HPR, we do anticipate the U3 stage to understand accident scenarios with spent fuel shuffling operations.

With U4, we expect to review the full gamut of technical areas as mentioned before with both scale and melt core.

MEMBER PETTI: So just so I understand, U3 you mean shuffling in core like we do in light water reactors?

MR. ALGAMA: Yes, sir.

MEMBER PETTI: Okay, okay.

MEMBER REMPE: And, Don, if you'll go back a slide? Okay. So this is why, and I think Dave captured it correctly by saying this is a bit different than the folks that are thinking about putting the core in the vessel or some container and installing the whole reactor vessel at the site.

And so perhaps this is one type of a heat pipe reactor, but there are other types where you have a fully loaded core that you move to the site. And that's not reflected in this diagram on your report, right?

MR. ALGAMA: Yes, ma'am. That's correct. When we started this work, we really looked at the designs that were being evaluated in Volume 3, and we used that to drive this report because we thought that was a good representation of what might come forth in the near future.

MEMBER PETTI: This is why I think a footnote to recognize that there are other options.

MR. ALGAMA: Yes, sir.

(Simultaneous speaking.)

MEMBER PETTI: -- if you can change the whole, you know, strategy of the report. But it's just that, you know, you could say, yes, we're aware of that other thing over there so.

MEMBER REMPE: So, yes, I think especially because I think Amy Cubbage mentioned this at a stakeholder meeting last month maybe, actually October or November, I forgot now which month. But she talked about that this might be a policy, challenge some policy issues. But it's something that the Agency needs to observe and note that they are aware of this, and they are starting to think about it.

MR. ALGAMA: Yes, ma'am. I'm going to go to the issues here. For this analysis, we will be using PBR 400 as in Volume 3.

This information is from NGNP, in other words that we know there are two types of HTGRs we can look at though in the form of pebble bed and prismatic type. The main difference between the two is expected to be with the fuel utilization stage, however, where the pebble bed design is not expected to have a U3 stage for fuel shuffling, used fuel handling inspection, et cetera.

For the PBR 400 though we expect what will drive this work from Volume 3, we expect about 400,000 pebbles each with tens of thousands of TRISO kernels within the reactor core, and helium is used as the coolant.

As far as the approach, this will look just like the FHR section that we just discussed. For MSRs, currently we're looking at the MSRE as the driver for this fuel cycle report.

Much reactor design information exists

along with models already developed within Volume 3 but not much involving the fuel cycle. This will have to be more of a research activity in which fuel salts can be transported to the site and diluted with salt available onsite before using in the reactor for example. More work needs to be done from a fuel cycle perspective.

As before, E1 and F1 are addressed elsewhere because there will be a UA6 initial phase. F1 fabrication step is looking at fabricating UA6 into uranium dissolved in salt in which fuel salt manufactured at F1 step is expected to be transported to the site where it would combine with fuel salt at the site and hydraulically transferred to the reactor circuit.

This stage will focus on actions that we're looking at criticality, chemistry use, et cetera there.

In the U1 step, we will look at criticality, shielding and issues and operations such as blending, handling, et cetera.

And in the U2 stage, power production, unlike chemical processes, will be covered in Volume 3. But refueling and processing capabilities are expected to be needed to remove salt and extract fission gas during operations. So that might be covered in this report and in Volume 3 as appropriate.

In the U4 stage, effort will be spent at criticality issue being regular transport and other chemical processes of interest that we identify.

This work has all other areas that we intend to make use of. From the front end UA6 works for the ATF inherent work. There are commonalities, and we will leverage those as much as we can.

So Volume 3 we will leverage the reference designs developed there and companion work to understand nuclear data -- and companion work that is being utilized to understand nuclear data performance.

This is useful as this not only helps define the fuel cycle for what we're going through but the radio fuel characteristics that drive the back end.

In the implementation phase, we also are intending on expanding collaboration with the DWD re-programs that are in this area upon the start of the work.

We are aware the DOE expects a certain amount of time looking at various fuel cycles, the efficacy of the fuel cycles and a number of reactor designs.

In conclusion, being we had a reasonable approach, a reasonable strategy in the reference to delta strategy benchmarked against the LWR fuel cycle, we believe that the development assessment work being performed under Volume 3 will help cover the development needs in Volume 5 so we don't expect new phenomena that aren't already captured in our codes.

What we're mainly focusing on is understanding how to revalidate our codes and what does that mean when we have more or less or in between months of validation data, whether we can mitigate the lack of data by using new methods and where we will just have to have new data available.

We believe that sufficient experience in the application of SCALE and MELCOR to non-reactors exists to start the process. But this experience will be developed and refined as we get more experience and implementation and also from DOE industry.

We will leverage other NRC programs to the extent possible, including Volumes 3 and 4 as the scenario dictates. That's all I have today. Thank you.

CHAIR BLEY: Thanks. Kim, do you have anything more?

MS. WEBBER: No. Not at this time,

Dennis, not specifically.

CHAIR BLEY: Members, if you have any questions, bring them up now please. After public comment, I'm going to go around and have everybody discuss a couple of things. But is there anybody on the committee who wants to ask any more questions at this point?

MEMBER PETTI: So, yes. I had one. I'm still struggling with after fabrication -- there is only one fabricator in the country today that can handle HALEU material that has a license from the NRC.

So this is, again, one of these assumption things. They already have a license. So they can do a lot of stuff, and it may not actually require, you know, an NRC review.

MR. ALGAMA: I see.

MEMBER PETTI: Because they have all of the, you know, safety paperwork in place.

It's probably worth talking about somewhere just, you know, what would have to happen to stand-up, you know, a fabrication plant that can handle HALEU. It's a lot different than LEU, you know, LWR fuel, whether that be modifying, you know, an LWR fuel vendor to allow them to handle HALEU or not so if someone wants to get into the game, you know, brand new.

MR. ALGAMA: Would this be an extension to this work? The whole idea was to try and show core readiness with this --

MEMBER PETTI: Yes. To me, it's just a footnote so you guys recognize that there are different options. One is a current LWR fuel vendor wants to make these advance fuels or there is the one vendor who can handle up to HEU today or you got a brand new guy coming in that wants to do it all themselves.

MR. ALGAMA: Yes, sir.

MEMBER PETTI: And that how you would apply these tools would differ for each of those three options, you know, just because of where they are in their licensing basis.

MR. ALGAMA: Understood.

CHAIR BLEY: Thanks. Anybody else?

MS. WEBBER: That's a good comment though, Dave. Thanks for that.

MEMBER PETTI: Okay. I mean, one of the things that just it struck me was all of this criticality analysis. Just so you guys are aware, the coaters, where you put the coatings on the particles are critically safe. They're designed to be critically safe.

So these guys, you know, this is their business, the people who fabricate. They're well aware of all of the rules and incorporating the safety, you know, into the designs of their system.

I think it's more difficult when we start talking about the fast reactor fuel, you know, who is going to step forward as an industrial supplier is more difficult. I haven't seen anything, you know, because for years it's just been done, you know, so some, say mom and pop at INL, for the EBI2 core really hasn't been done after that in any large scale.

MR. ALGAMA: Yes. I think it's important to understand we're not trying to redo or generate new safety items of interest. We're just trying to find a sufficient number of scenarios that we could test our codes, I think, just so I'm clear. The intention was not to actually do a review. Does that help or?

MEMBER PETTI: Yes, I mean, maybe, again, maybe making that clear may be --

(Simultaneous speaking.)

MEMBER PETTI: -- if it isn't clear enough because that didn't jump out at me, I guess.

MR. ALGAMA: Yes, sir. We can make it clear. And doing a full blown review would be a much bigger task that I wasn't anticipating so.

MEMBER PETTI: Right.

MS. WEBBER: And I think overall, you know, so I reflect on the number of comments related to, you know, scenarios given the breadth of, you know, advance reactor designs. And I think, you know, what common in many of the comments is that we really need to include a set of scenarios, fuel cycle scenarios that will -- I hate to use the word bound, but a set of fuel cycle scenarios that will cover most of what we would anticipate.

MR. ALGAMA: Originally, the idea was to do that in the implementation phase. But we can try to hypothesize something up-front but that might change when we start to actually do the work. Is that okay?

(Simultaneous speaking.)

MR. ALGAMA: I'm sorry. Go ahead.

CHAIR BLEY: What I worry about that is if you do it partially now, we've got to make it real clear that it's got to be revisited in substantial detail whether --

MR. ALGAMA: Yes, sir.

CHAIR BLEY: That's the only answer that I would have.

MR. ALGAMA: We originally thought of giving some more examples of what we would look at. But because of that fear, we decided to keep it just as a plan and then really drill down into it when we implement. But we can try to come up with some compromise approach that makes sense, that provides clarity, if that helps.

MS. WEBBER: Well, and I think to -- maybe Dennis, this was your question or maybe it was Joy's question about updating the reports. I mean, this Volume 5 conceivably may be one where given the knowledge that we have today and the uncertainties about where, you know, the fuel cycle technologies are going in the future, especially for the further out, you know, design concepts, this volume may be one that we, you know, note that an update would be necessary potentially.

But, you know, I see this document as really providing the strategy. Right now, it contains notionally 10 reports. And, you know, 10 reports and each report represents, you know, a look at that fuel cycle with the identification of gaps and methodologies to close the gaps and, you know, updates to the codes and things like that. But, you know, as we learn more then it may become a set of not only 10 but a few others.

MR. ALGAMA: Yes, ma'am. It could be bigger or smaller.

MS. WEBBER: Right.

CHAIR BLEY: That all seems reasonable to me. I WOULD point out to you that although the discussion was about reactors, it applies equally well to fuel cycles.

We had a lessons learned letter report recently, a couple other of our letter reports. And in a recent meeting -- actually, I'll go with the OMB, Mr. Fleming, with the group putting together the guidance, where he identified a series of reports in the same vein that lay out approaches to search for initiating events and scenarios for problems.

You know, this is people's business where, yes, they're doing it well. But you've got to really do a thorough search to find the things that will surprise or there will be surprises later. So there's some hope for that if you look at those recent references.

MR. ALGAMA: Yes, sir. Thank you. You said inside the LMP? I'm sorry.

MS. WEBBER: I was going to say, Don, maybe that's something we can talk to Derek and whoever offline to figure out what those resources are because off the top of my head it doesn't ring a bell.

CHAIR BLEY: We can do that. We'll also talk about -- the meeting will be in February. We're on February 20, 21 for Volumes 4 and 5. So we'll have an admin call set up to talk about some of that, and we can give you some of that other information.

Anything else from the members? I'm going to go around for public comments and then we'll come back.

MEMBER REMPE: Dennis, I guess, again, I would point out that as one searches for initiating events, I think a review of history and root causes for events in the past and what it considers more recent events as well as some of the non-LWR experience in the U.S. where DOE backed the Atomic Energy Commission days where they were the developer as well as the regulator offers some really good lessons in thinking about what needs to be considered here.

MR. ALGAMA: Understood.

CHAIR BLEY: Can we get the tone line open for comments?

MR. DASHIELL: The public bridge line is open for comments.

CHAIR BLEY: Thank you. Is there anyone in the public who would like to make a comment? If so, please state your name and make your comment at this time. Going, going. Okay. We can close the bridge line.

Instead of going around to all the members, the intention is to have the meeting in February to write a letter report on Volumes 4 and 5. And I want to divert for just a second back to Kim. Kim, you expressed that you guys didn't have an interest in revisiting the changes to Volumes 1, 2 and 3 in the overview report.

But I don't know if it fell through the cracks, or crack, because of COVID or if there's other reasons, but we have never received any real response letter on our letter on Volumes 1, 2 and 3. So given that we hadn't --

MS. WEBBER: Actually, I have that. I think I have that because I think we crafted it. But I think we can try to dredge that up.

CHAIR BLEY: That might take care of any revisiting them in February. So if you can find that and get it in the system, we'll talk about that, too, when we put them up. I'd like to revisit those because so far we don't have anything from you officially.

MS. WEBBER: Okay. Yes. I'll see if I can resurrect that. But I think I recall, you know, there was a specific ticket with a response.

CHAIR BLEY: And it never made it up on the NRC website either, it's normally there.

MS. WEBBER: Okay.

CHAIR BLEY: So the intention is to write a letter on Volumes 4 and 5 and maybe it's something about dealing with our previous recommendations from November of last year.

Are there any members of the subcommittee at this time who would like to comment specifically? Instead of going all around the room, I'll just ask you to come forward. Mike Corradini, anything from you as our consultant?

MR. WIDMAYER: Hey, Dennis, this is Derek. Mike's currently out of the meeting.

CHAIR BLEY: Oh, okay. He said he might not be here. I saw him so I screwed up one. Okay. So without any further comments, we'll look forward to getting together in February to talk about Volumes 4 and 5. We'll have that offline meeting with Kim and maybe some others before then. So at this time, we are adjourned.

MR. ALGAMA: Yes, sir. Thank you very much.

MS. WEBBER: Yes. Hey, Dennis, is there a date for that fall committee meeting?

CHAIR BLEY: Oh, geez, Derek? Yes, it's in February.

MR. WIDMAYER: Yes. We have dates but we haven't done an agenda or anything yet but.

CHAIR BLEY: We don't have it pinned down. It will be the 4th or the 5th.

MR. WIDMAYER: Yes.

MS. WEBBER: Oh, okay. That's good enough for now.

MR. WIDMAYER: Yes.

MS. WEBBER: Okay. Right. Well, I do appreciate you all taking the time and putting some really good thoughts together about how to improve not only the strategy but the quality of the report. And I just really appreciate your time. I know you're busy, and there's a lot going on. So thank you very much.

MR. LEE: This is Richard Lee. I want to make a comment.

MS. WEBBER: Okay.

CHAIR BLEY: Okay. I guess we can reopen

and take your comment.

MR. LEE: In response to Dennis, I mean, Dave Petti about the fast reactor fuel fabrication, our staff can reach out to the French and the Japanese to learn what they have done with respect to the fast reactor stuff so.

MEMBER PETTI: Yes. But, Richard, that's oxide fuel. And the U.S. is the only ones who make the metal fuel.

MR. LEE: Yes, but the thing is that you are worried about mostly, like, the enrichment aspect of it. So there may be some applicability from those.

MEMBER PETTI: That's true, yes.

MR. LEE: Yes.

MEMBER KIRCHNER: Yes, that part might be. But as Dave points out -- this is Walt Kirchner. Yes, their experience is mainly oxide. We had at that TF oxide fuel. But the concepts that we see coming seem to be leaning towards using the metallic fuel, which is the argon INL EBR-II experience.

MR. LEE: Let us remember if I'm going to validate the neutronics aspect of it, I can use a lot of different forms in terms of criticality so. The physics is still there with fast spectrum behavior for the uranium aspect of it.

MR. MOORE: Chairman Bley, this is Scott Moore. Can I be recognized?

CHAIR BLEY: Yes, you may, Scott. Go ahead.

MR. MOORE: To follow-up on the conversation, the full committee meeting in February is on February 4 and 5. And as Derek mentioned, it does not yet have an agenda.

The second thing is just to note that Steve Schultz is also in the meeting or at least the list of attendees is showing Steve, our consultant on.

CHAIR BLEY: Thank you very much.

MS. CUBBAGE: Dr. Bley, this is Amy Cubbage. May I be recognized?

CHAIR BLEY: Who is this?

MS. CUBBAGE: Amy Cubbage.

CHAIR BLEY: Yes, Amy.

MS. CUBBAGE: Yes, I just wanted to note that the staff contracted with the national labs to look at the safety and hazards associated with fuel fabrication in the reports available on the NRC website, including specifically a metal fuel fabrication safety hazards report.

CHAIR BLEY: Thank you. And that's publicly available now?

MS. CUBBAGE: Yes, it is. I can provide the link to Derek.

CHAIR BLEY: Thank you. That will be helpful. Well, we sort of reopened the meeting. I think I heard Joy.

MS. WEBBER: No, it was Kim. Amy, can you copy me on that, too?

MS. CUBBAGE: Absolutely.

MS. WEBBER: Thank you.

CHAIR BLEY: Anybody else? We're finishing way early. I already thought we were adjourned once, but I'll give you another minute here.

Okay. If nothing more, we will adjourn at this time for real. And we'll see you again in February. Thanks to all.

MS. WEBBER: Thank you all. Happy Holidays.

CHAIR BLEY: Happy holidays. Bye-bye.

MR. ALGAMA: Thank you. Goodbye.

(Whereupon, the above-entitled matter went off the record at 10:59 a.m.)

Implementation Action Plan (IAP)
Strategy 2 – Volume 5

Code Application Plans for Advanced Reactor Nuclear Fuel Cycles

December 1, 2020

Kimberly A. Webber, Ph.D.
Division of Systems Analysis
Office of Nuclear Regulatory Research

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Agenda

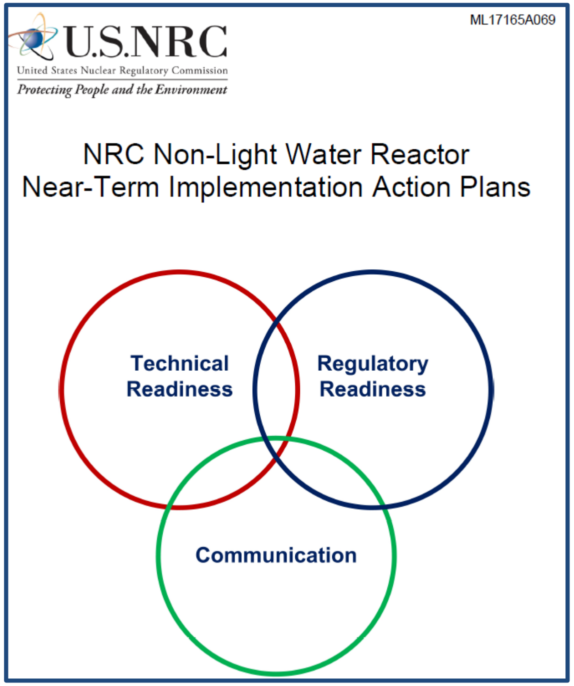
- Staff Introduction
- IAP Strategy 2 Overview
- ACRS Strategy 2 Meeting Schedule
- Non-LWR Fuel Cycle Analysis Plan (Vol. 5)
  - Overview of Existing Fuel Cycle and Analysis
  - Advanced Reactor Fuel Cycle and Analysis
  - Leveraged Programs
  - Concluding Remarks

NRC's "Be Ready" Attitude

- Improve mission value while enabling safe operations
  - Deliver cost savings
  - Develop regulatory tools
  - Leverage collaborations
  - Build staff expertise

BlueCRAB

NRC's Integrated Action Plan (IAP) for Advanced Reactors



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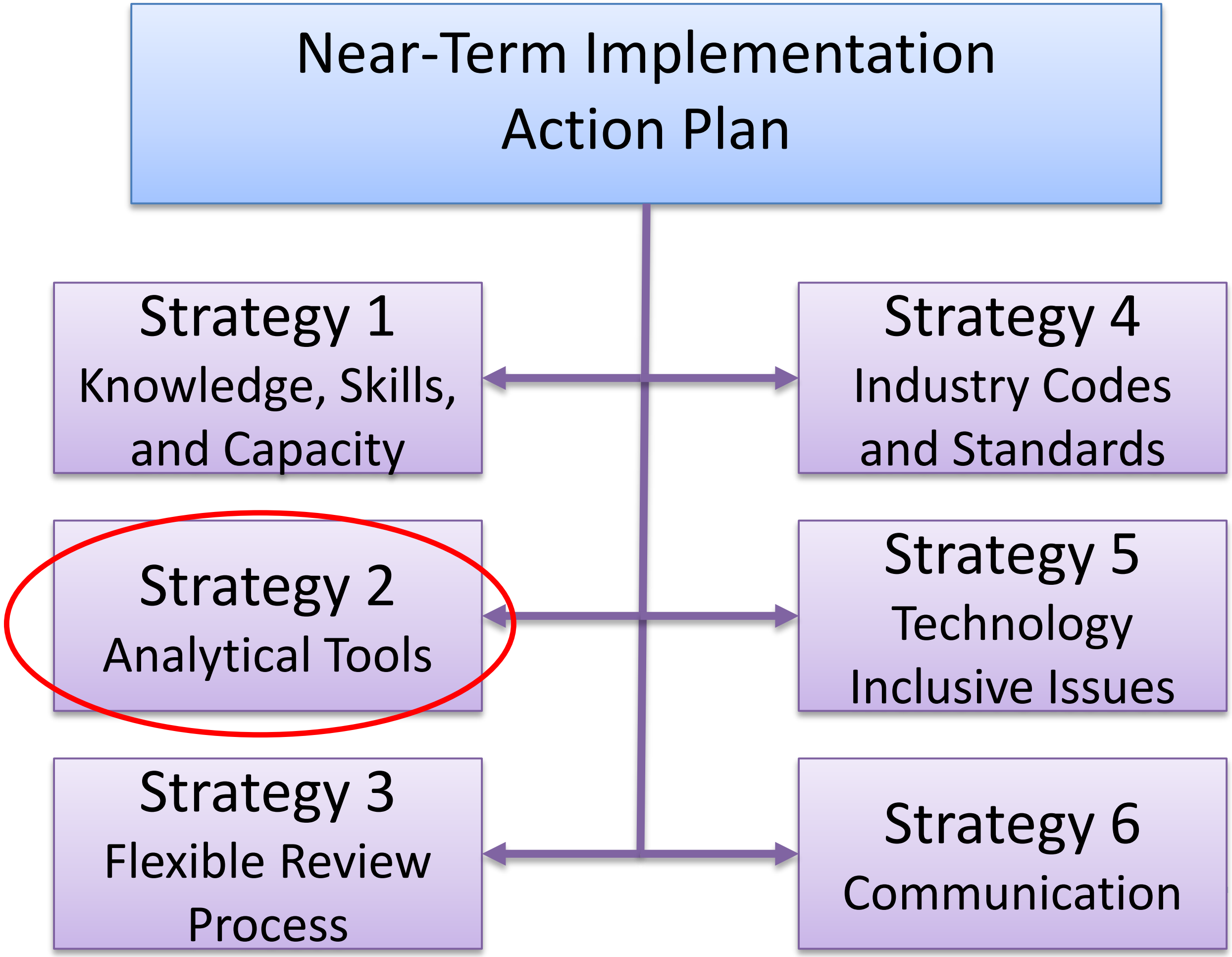
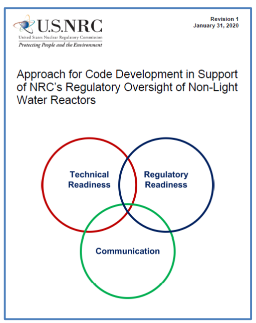


Strategy 2: Computer Code Readiness
Code Development Plans

These Volumes outline the specific analytical tools to enable independent analysis of non-LWRs, "gaps" in code capabilities and data, V&V needs and code development tasks.


U.S.NRC
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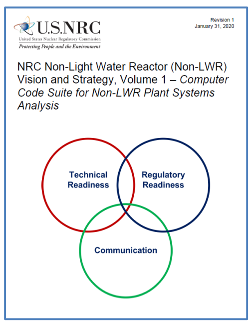
Revision 1
January 31, 2020
Approach for Code Development in Support of NRC's Regulatory Oversight of Non-Light Water Reactors

Introduction
ML20030A174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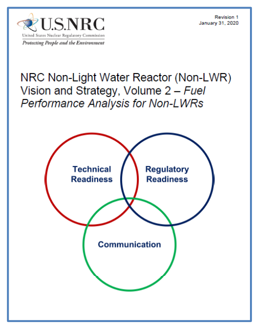
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 1 – *Computer Code Suite for Non-LWR Plant Systems Analysis*

Volume 1
ML20030A176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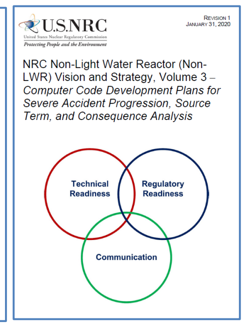
January 31, 2020
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 2 – *Fuel Performance Analysis for Non-LWRs*

Volume 2
ML20030A177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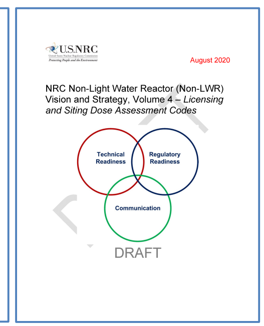
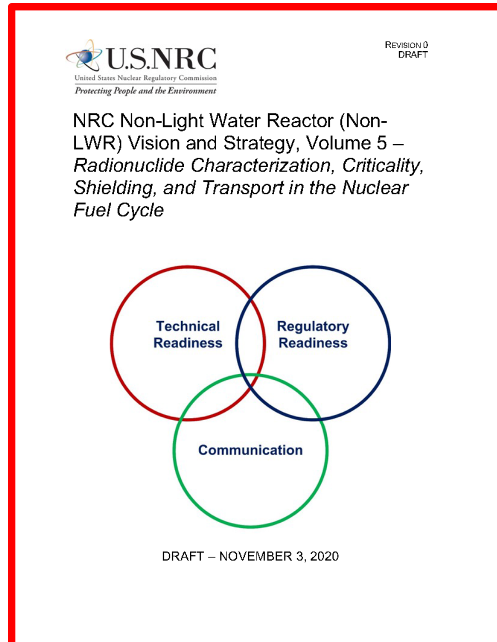
January 31, 2020
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

Volume 3
ML20030A178


U.S.NRC
August 2020
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 4 – *Licensing and Siting Dose Assessment Codes*

Volume 4
ML20028F255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Revision 0
DRAFT
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

DRAFT – NOVEMBER 3, 2020

Volume 5
ML20308A744

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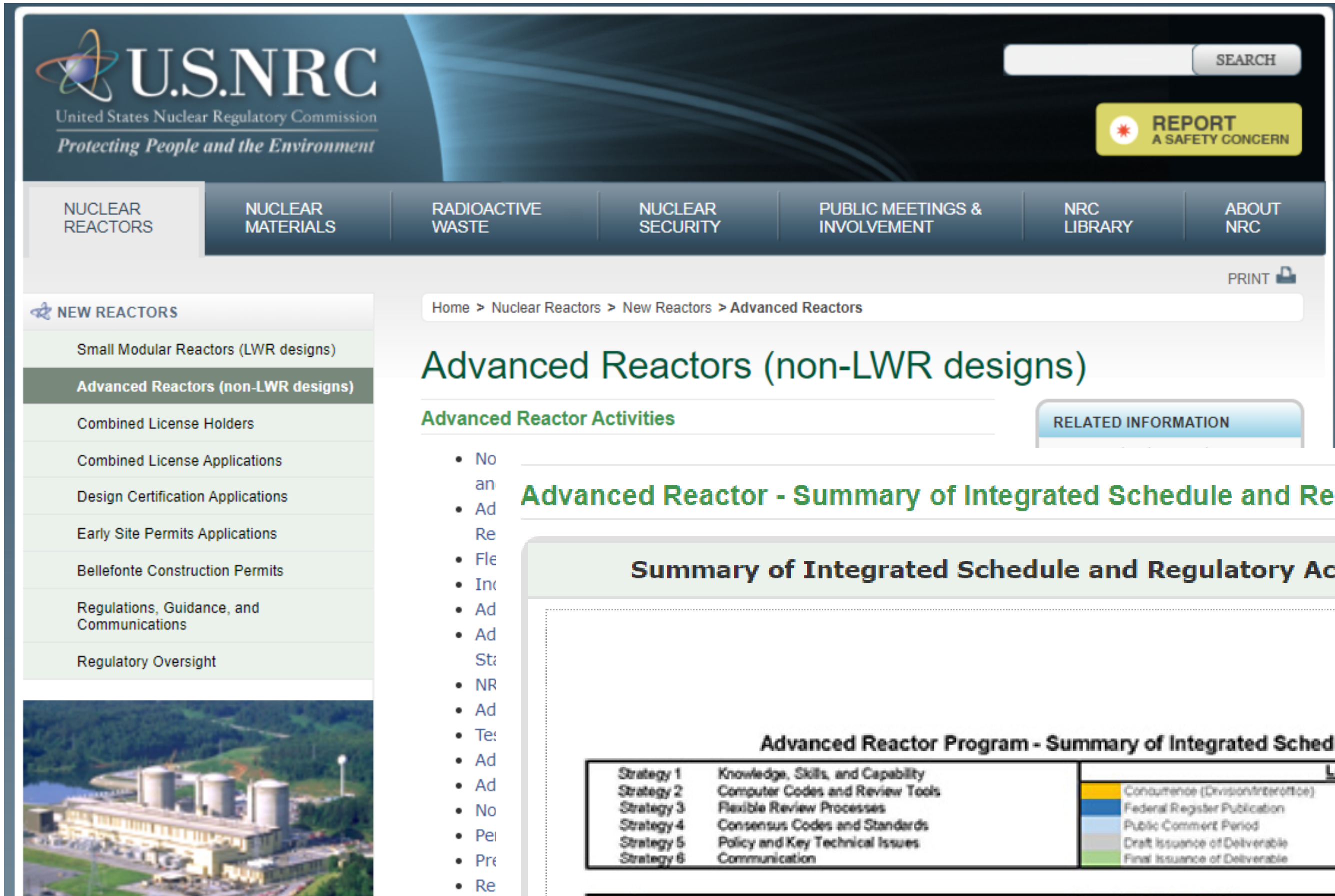
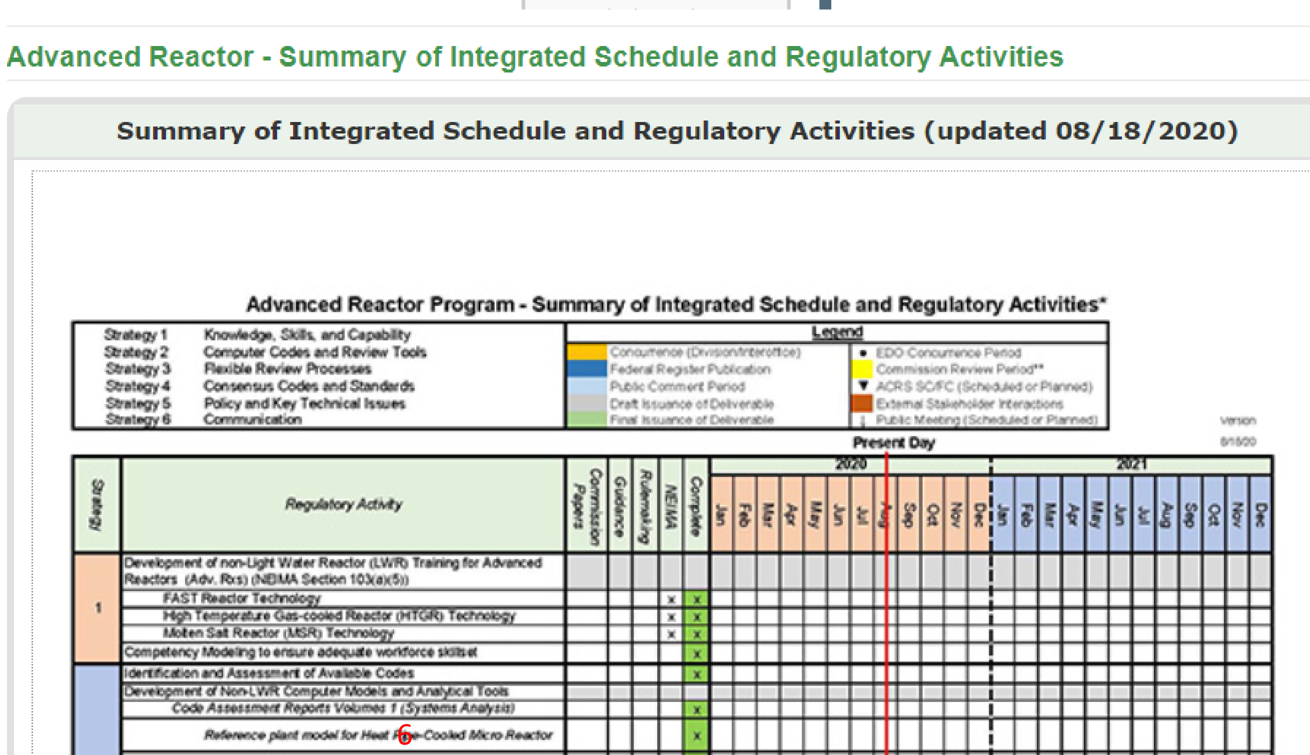
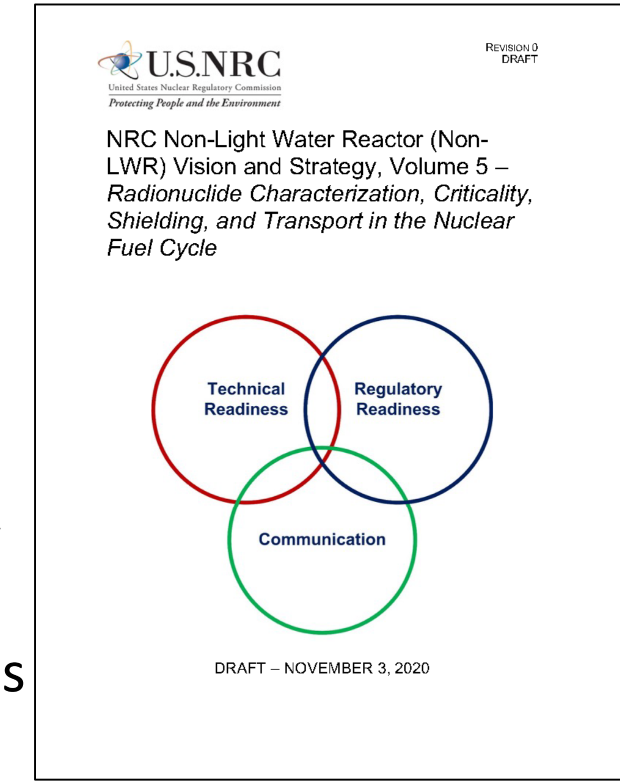
NRC's Integrated Action Plan (IAP) Status





Overview of Volume 5

- Assessment and use of existing NRC computational tools for accident analysis (Volume 3) and consequences (Volumes 3/4)
- Incremental development approach based on existing LWR fuel cycle as reference
- Staff experience with anticipated non-LWR fuel cycle and use of computer codes
- Development of non-LWR fuel cycle reports and publicly available input decks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REVISION 0
DRAFT

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

DRAFT – NOVEMBER 3, 2020

Volume 5
ML20308A744

“NRC non-Light Water Reactor Vision and Strategy, Volume 5: Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle”

Presented by
Don Algama (RES) and Drew Barto (NMSS)

United States Nuclear Regulatory Commission
Office of Nuclear Regulatory Research (RES)
Nuclear Materials Safety and Safeguards (NMSS)

Acknowledgements

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- Dr. David Luxat (Sandia) and Dr. William Wieselquist (ORNL) were instrumental in the plan development.

IAP Strategy 2 Volumes to Date

Revision 1
January 31, 2020

Approach for Code Development in Support of NRC's Regulatory Oversight of Non-Light Water Reactors

Technical Readiness | Regulatory Readiness | Communication

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*

Technical Readiness | Regulatory Readiness | Communication

Volume 1
[ML20030A176](ML20030A176)

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy - Staff Report: Near-Term Implementation Action Plans

Technical Readiness | Regulatory Readiness | Communication

DRAFT

Volume 2
Volume 2 – Detailed Information
[ML20030A177](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*

Technical Readiness | Regulatory Readiness | Communication

Volume 3
[ML20030A178](ML20030A178)

August 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 4 – *Licensing and Siting Dose Assessment Codes*

Technical Readiness | Regulatory Readiness | Communication

DRAFT

Volume 4
[ML20028F255](ML20028F255)

Revision 0
DRAFT

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

Technical Readiness | Regulatory Readiness | Communication

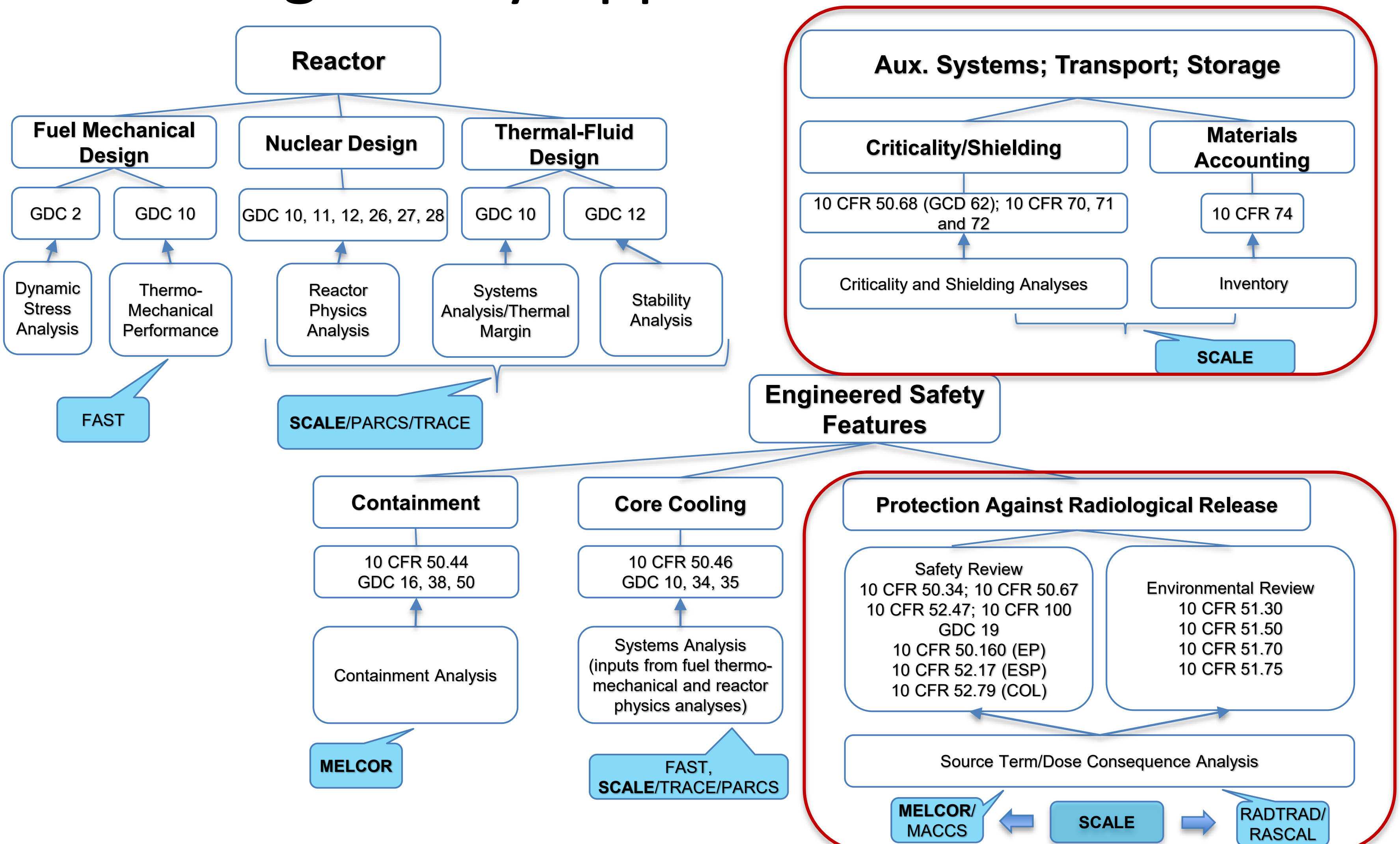
DRAFT – NOVEMBER 3, 2020

Volume 5
ML20308A744

Objectives

- Elements of the fuel cycle plan
  - Demonstrate computer code readiness
  - Assessment and use of existing NRC computational tools for accident analysis (Volume 3) and consequences (Volumes 3/4)
  - Incremental development approach based on existing LWR fuel cycle as reference
  - Staff experience with anticipated non-LWR fuel cycle and use of computer codes
  - Development of non-LWR fuel cycle reports and publicly available input decks

Regulatory Application of Codes



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Transportation and Storage Licensing (LWR)

analysis end-points

ENDF/B
Physics data
Thermal scattering law, resonance data, energy distributions, fission yields, decay constants, etc.

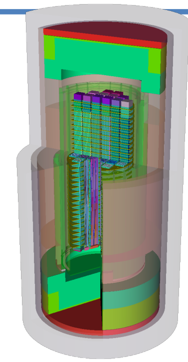
JEFF Activation
Isomeric cross sections, activation reactions

Sources4C
neutron emission data (alpha,n)

ICRP
dose conversion factors, radiotoxicity

NIST
natural abundance, atomic mass

scale
Nuclear Systems Modeling & Simulation



AMPX
Validated cross section libraries in multigroup (O(100g)) or continuous-energy (O(100,000g); depletion and decay data

SHIFT/MAVRIC
3D shielding and dose rate analysis

ORIGEN
General depletion, decay, source term

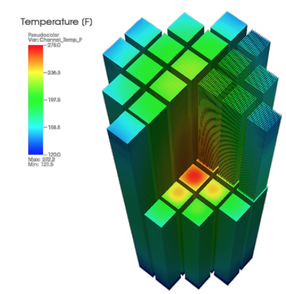
CSAS
3D criticality safety analysis

TRITON/SHIFT
General reactor fuel neutron transport + depletion

ORIGAMI
Reactor-specific radioactive isotopics/source term characterization

Thermal Analysis

Structural and containment (cask) analyses



Severe Accident & Consequence Analysis (LWR/non-LWR example)

analysis end-points

ENDF/B
Physics data
Thermal scattering law, resonance data, energy distributions, fission yields, decay constants, etc.

JEFF Activation
Isomeric cross sections, activation reactions

Sources4C
neutron emission data (alpha,n)

Kinetics Data
nuclide-specific beta-effective, precursor data

NIST
natural abundance, atomic mass

scale
Nuclear Systems Modeling & Simulation

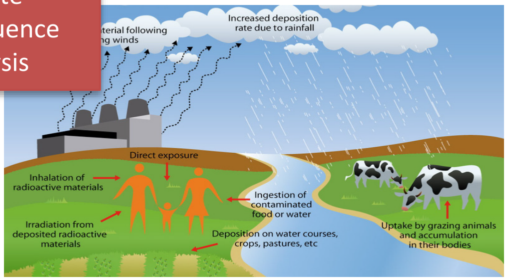
AMPX
Validated cross section libraries in multigroup (O(100g)) or continuous-energy (O(100,000g); depletion and decay data

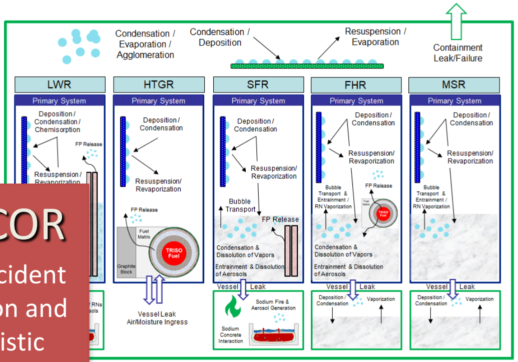
ORIGEN
General depletion, decay, source term

TRITON/SHIFT
General reactor fuel neutron transport + depletion

ORIGAMI
Reactor-specific radioactive isotopics/source term characterization

MACCS
Offsite consequence analysis





MELCOR
Severe accident progression and mechanistic source terms

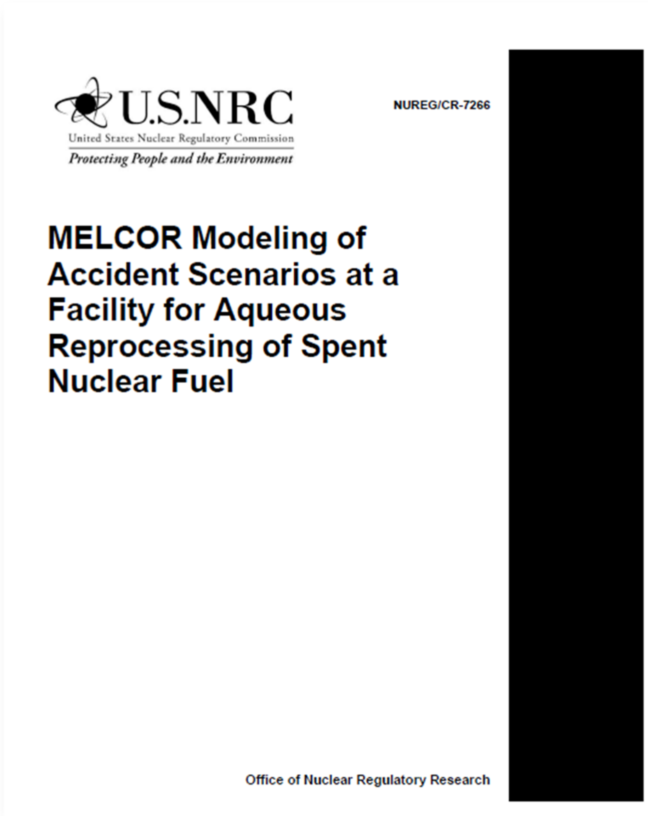
"NRC Non-Light Water Reactor Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression , Source Term, and Consequence Analysis," Revision 1, January 2020, ML20030A178

Examples of Existing Fuel Cycle Analysis

- Level 3 PRA Project
  - SCALE/MELCOR are used to support PRA development of accident sequences and source terms including non-reactor scenarios for the spent fuel pool
- NUREG-2161
  - SCALE/MELCOR was used to study the performance of a SFP under severe accident conditions
- NUREG/CR-7108/7109
  - Here SCALE was used to estimate isotopic depletion and criticality code, and cross section data bias related to burnup credit in spent fuel storage and transportation systems

Examples of Existing Fuel Cycle Analysis

- Barnwell – Non-Reactor Safety Assessment
- SCALE/MELCOR utilized as part of best-estimate analysis methodology in NUREG/CR-7266
- Spent fuel inventories developed in SCALE package
- Aerosol transport modeling
  - Integral analyses estimate radiological transport and release
  - Aerosol modeling enables estimation of transport of hazardous material within facility and to environment
- Accident scenarios considered relevant to broad range of facility accidents
  - Explosion scenario
  - Fire scenario
  - Combined explosion and fire scenario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NUREG/CR-7266

MELCOR Modeling of Accident Scenarios at a Facility for Aqueous Reprocessing of Spent Nuclear Fuel

Office of Nuclear Regulatory Research

non-LWR Characteristics

Table 1-1. Comparison Between LWR and Non-LWR

Reactor Type	Enrichment (wt.%)	Fuel Form	Typical Discharge Burnup	Fuel Residence Time	On-Site Fuel Processing	Fuel Storage / Transport
LWR (Ref.)	<5	U Oxide	Peak Rod Average: <62 GWd/MTU Max Assembly Average: <55 GWd/MTU	Assemblies burned for approximately 3 to 4 cycles	No	Storage: Fresh and spent fuel storage on-site or off-site Transport: FE: UF_6 solid transport in 30B cylinders, fresh fuel assembly and fuel component (UO_2 powder/pellet) transportation packages BE: Used fuel transport and dry storage containers
LWR: HALEU /HBU (Ref.)	5 – 10	U Oxide	Peak Rod Average: ~75 Wd/MTU Max Assembly Average: ~60-70 GWd/MTU	Assemblies burned for approximately 3 to 4 cycles	No	
HPR	5 – 20	U Oxide U Metal	2-10 GWd/MTU	Up to 7yrs	No	To be evaluated*
SFR	5 – 20	U Metal	Up to 300 GWd/MTU	To be evaluated*	No	To be evaluated*
HTGR	5 – 20	TRISO (UCO or UO2) in pebble bed or prismatic array	100-200 GWd/MTU	To be evaluated*	No	To be evaluated*
FHR	5 – 20	TRISO (UCO or UO2) in pebble bed	100-200 GWd/MTU	To be evaluated*	No	To be evaluated*
MSR	5 – 20	^{235}U dissolved in molten salt	To be evaluated	2-3yrs	Yes	To be evaluated*

*Will be evaluated based on information available at the time work is undertaken, e.g. based on current DOE and industry input.

Analysis Approach

Develop accident scenarios by reviewing available information including documents such as:

- NUREG/CR-6410 "Nuclear Fuel Cycle Facility Accident Accident Analysis Handbook"
- NUREG-1520 "Standard Review Plan for Fuel Cycle Facilities License Applications"
- NUREG-2215 "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities – Final Report"
- NUREG-2216, "Standard Review Plan for Spent Fuel Transportation"
- DOE-HDBK-1224-2018: DOE Accident Analysis Handbook "Hazard and Accident Analysis Handbook"

Scope of Analysis

- Assess existing codes to cover neutronics and radionuclide and non-radionuclide hazards throughout non-LWR fuel cycles
- Consequence and radiation protection methods are covered under Volume 3/4
- Mining, milling, long term storage and disposal are not considered in this activity
- Leverage volume 3 non-LWR designs
  - Fluoride-Salt-Cooled (Solid-Fuel) High Temperature Reactor (FHR)
  - Heat Pipe Reactors (HPR)
  - Sodium Fast Reactor (SFR)
  - High Temperature Gas Reactor (HTGR)
  - Molten Salt Reactor (MSR)

Follow these analysis steps used in Volume 3 and previous fuel cycle work for LWRs

1. Define scenario
2. Identify safety related item(s) of interest
3. Ask the right safety questions / Phenomena of interest / Understand the dominant features
4. Survey experiments available that provide fundamental information
5. Develop physics models to capture dominant feature and allow prediction
6. Translate physics models into computer code
7. Perform verification testing (unit testing; and integrated testing as code complexity increases)
8. Perform validation with experiments. Capture the integrated codes performance (with uncertainty analysis)
9. Document findings

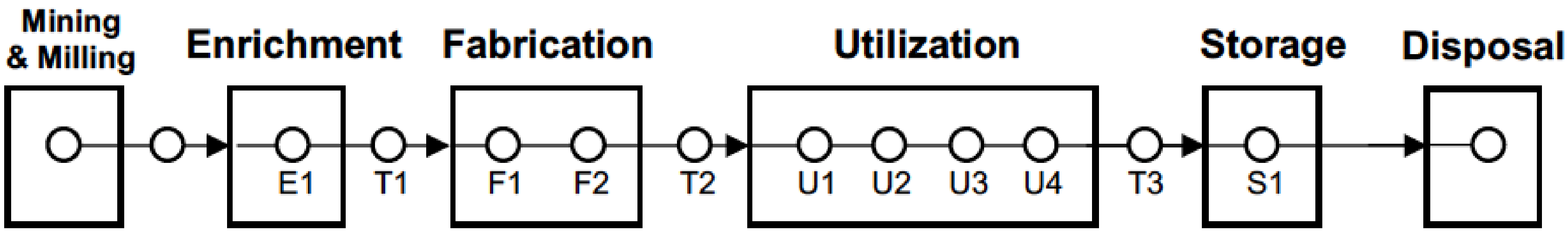
Deliverables

- 10 reports are defined as a result of this plan
  - Each report defines a set of accident scenarios during a portion of the fuel cycle
  - Perform assessment, analysis, and generate demonstration input files
- 5 non-LWRs currently considered and openly available reference designs defined in volume 3:
  1. FHR Fuel Cycle Analysis (Berkeley Mk. 1)
  2. HPR Fuel Cycle Analysis (INL Design A-MET)
  3. SFR Fuel Cycle Analysis (MET-1000/VTR)
  4. HTGR Fuel Cycle Analysis (PBMR-400)
  5. MSR Fuel Cycle Analysis (MSRE)
- 5 front end (FE) reports centralize FE analysis among these non-LWRs
  6. Enrichment and UF6 Handling up to 20 wt.%
  7. TRISO Fuel Kernel Fabrication
  8. Uranium Metallic Fuel Fabrication
  9. Fast Reactor Fuel Assembly Fabrication
  10. Pebble TRISO Fuel Fabrication

This organization of deliverables allows prioritizing specific designs and reducing overlap. For example:

- HTGR analysis requires the following reports 6→7→10→4.
- For FHR, it would require 6→7→10→1. 6,7, and 10 are already available!

Reference - LWR Cycle

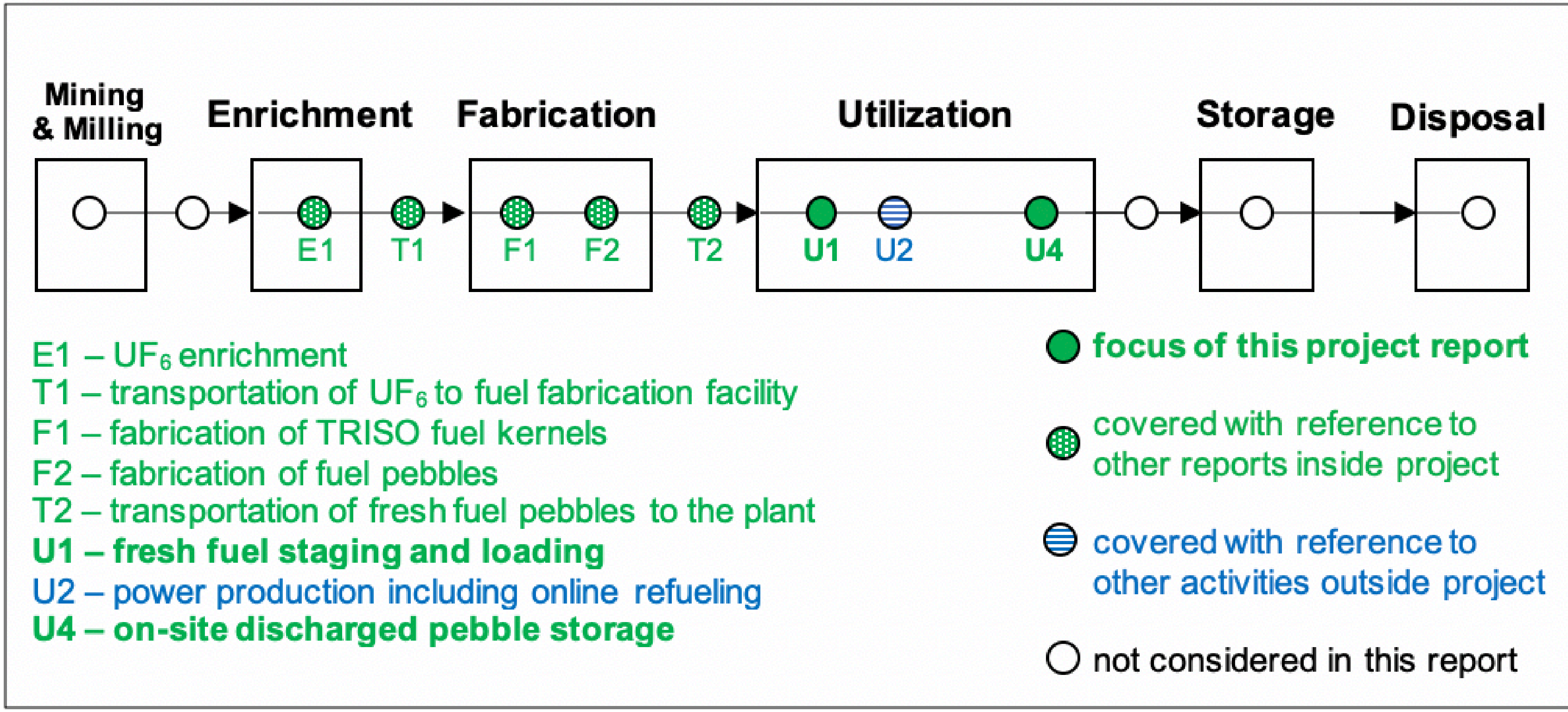


E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of UO_2 fuel pellets
F2 – fabrication of LWR fuel assemblies
T2 – transportation of fresh fuel assemblies to the plant
U1 – fresh fuel staging and loading
U2 – power production
U3 – spent fuel pool/shuffle operations
U4 – on-site dry cask storage
T3 – transportation of spent fuel to off-site storage
S1 – off-site storage

Each analysis report tackles one or more of the equivalent fuel cycle stages for each non-LWR.

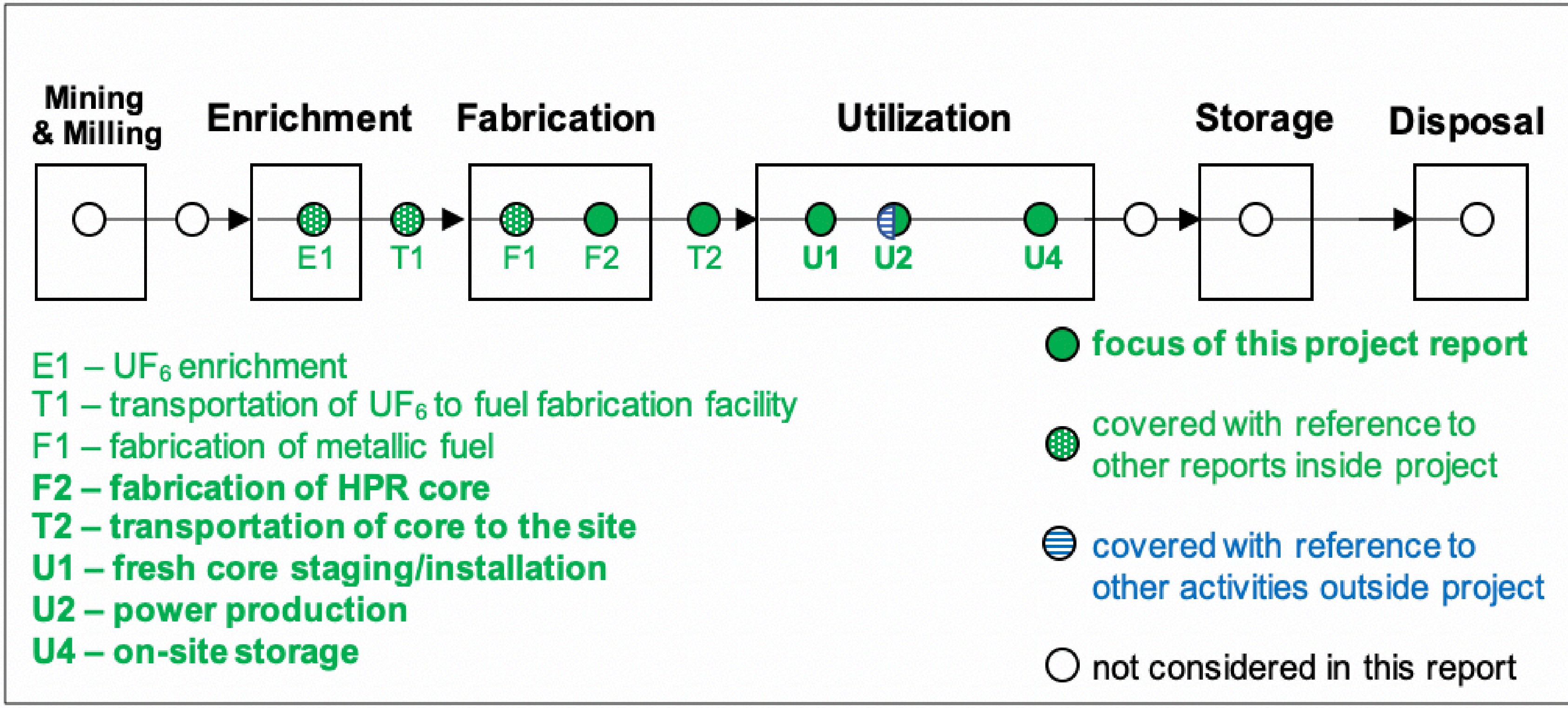
NOTE: Transportation off-site and off-site storage (T3 and S1) are currently not considered in this fuel cycle assessment plan due to uncertainty with this part of the back end.

FHR Fuel Cycle Report



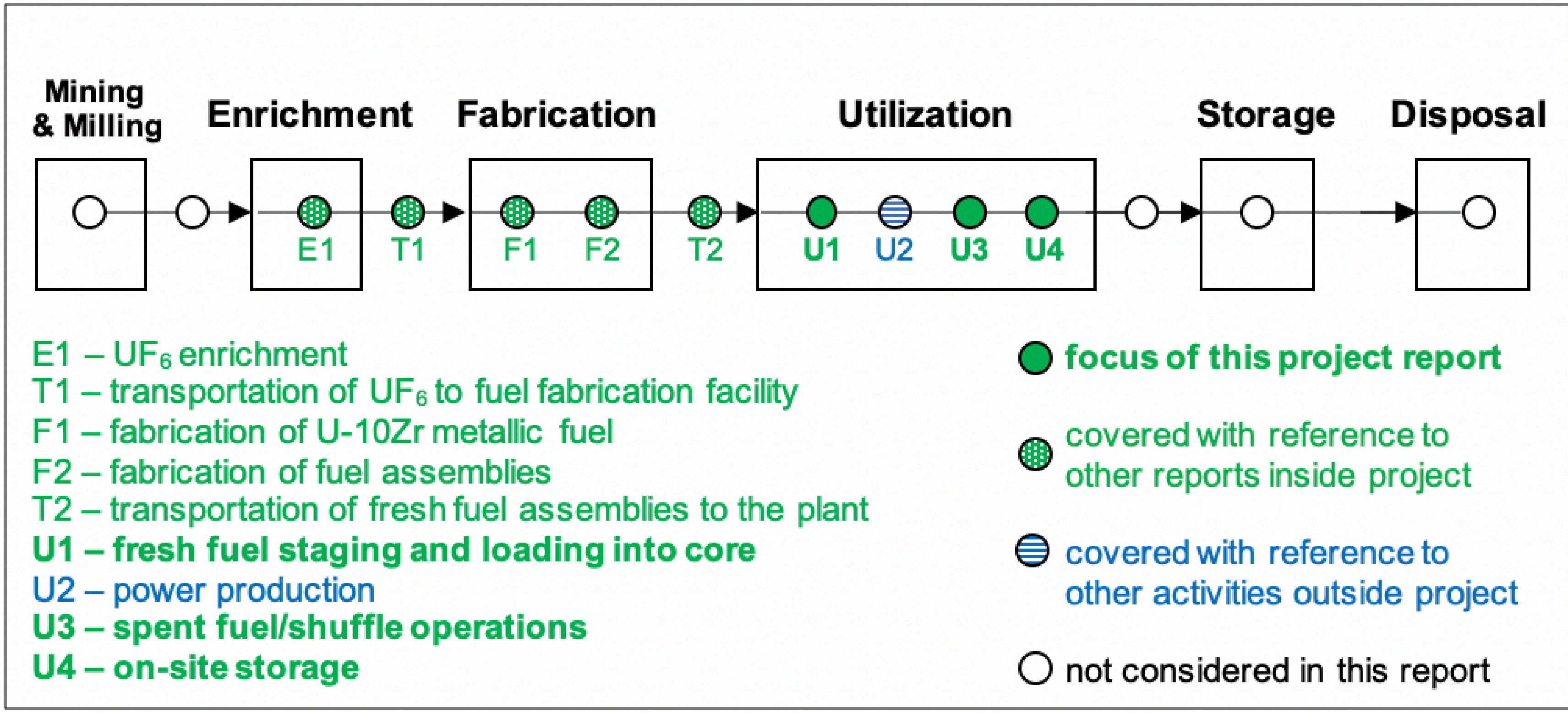
The FHR fuel cycle report develops and analyzes new accident scenarios related to stages U1 and U4 and links them to earlier front-end stages (E1, T1, F1, F2, T2) analyzed in this project and in-reactor scenarios U2 from volume 3.

HPR Fuel Cycle Report



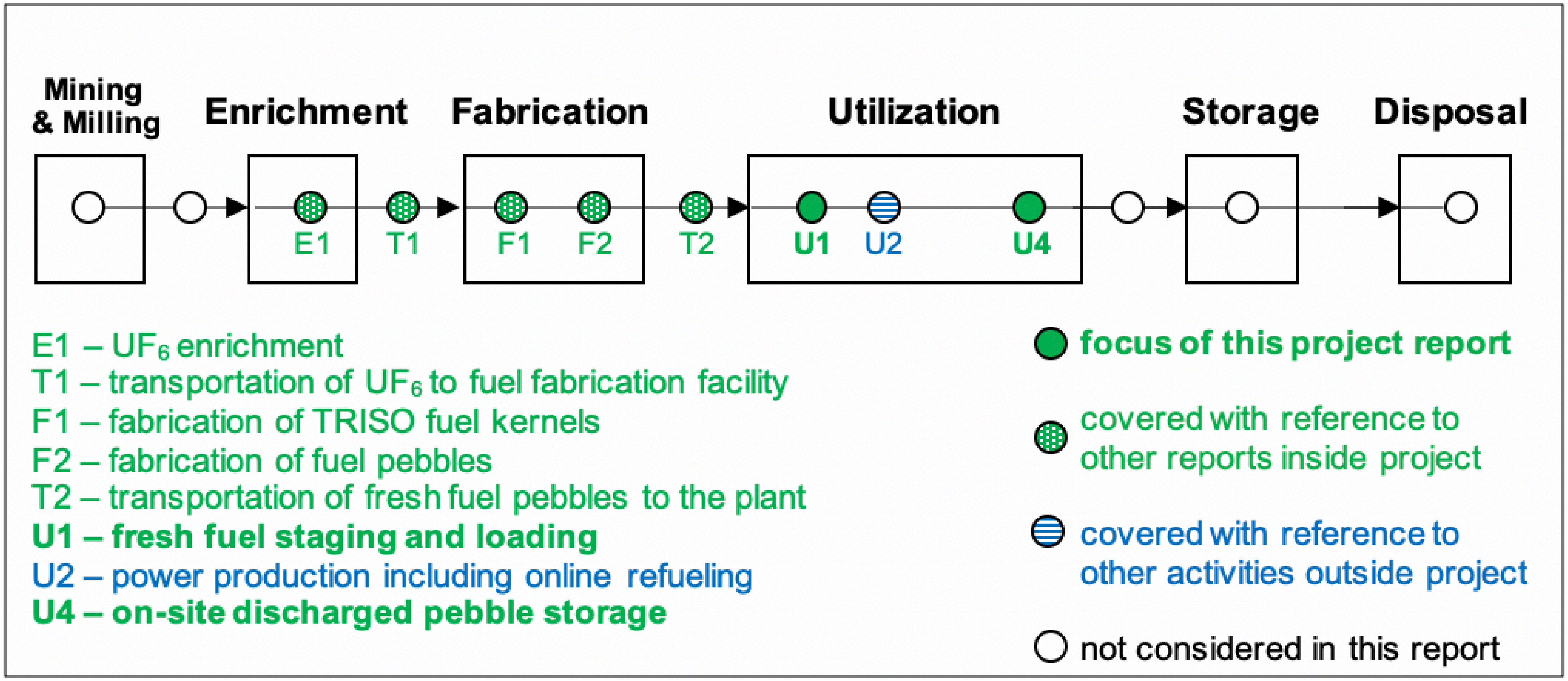
The HPR fuel cycle report develops and analyzes new accident scenarios related to stages F2, T2, U1 and U4 but also requires re-analysis of U2 for a metallic fuel system (current source term demo calcs using oxidic fuel). NOTE: The F2 and T2 front end stages are included in this report because fabrication and transportation of an HPR core will be specific to that design and thus nothing is gained from putting those stages in their own analysis reports.

SFR Fuel Cycle Report



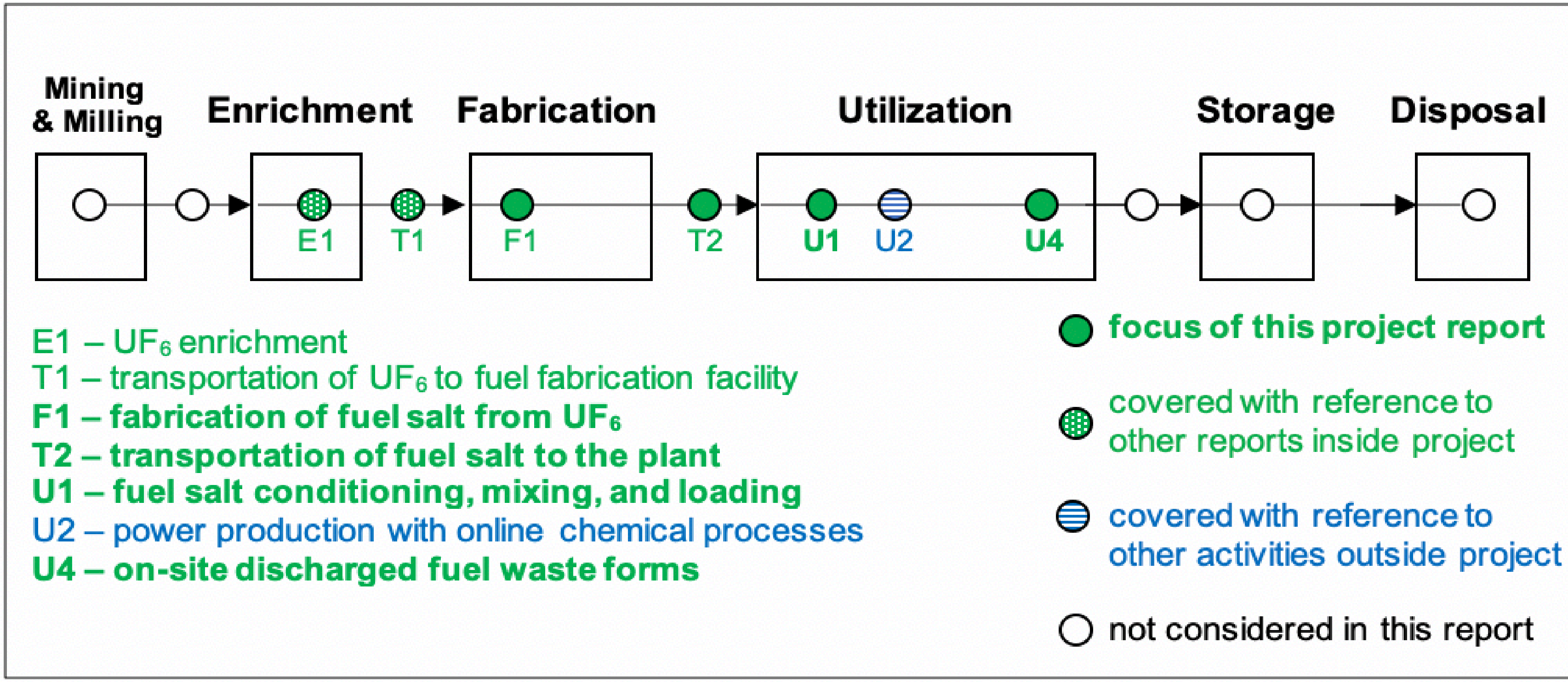
The SFR fuel cycle report develops and analyzes new accident scenarios related to stages U1, U3, and U4 and links them to previously studied E1, T1, F1, F2, and T2. NOTE: The F2 and T2 front end stages are their own report not because of overlap included in this report because fabrication and transportation of an HPR core will be specific to that design and thus nothing is gained from putting those stages in their own analysis reports.

HTGR Fuel Cycle Report



The HTGR fuel cycle report develops and analyzes new accident scenarios related to stages U1 and U4 and links them to front-end stages (E1, T1, F1, F2, T2) analyzed in this project and in-reactor accident scenarios U2 from volume 3. Front end analysis is basically the same as for FHR.

MSR Fuel Cycle Report



The MSR fuel cycle report has the least overlap with any other design and develops and analyzes new accident scenarios for F1, T2, U1, and U4 in the main MSR analysis and links them only to front end E1 and T1 for UF6 enrichment and transportation.

Leveraged Programs

- HALEU
  - UF_6 transport packages
  - Fresh fuel transport packages
- Volume 3 (codes and plant models)
  - Capabilities to characterize utilization stage
  - Hazardous material transport for non-water systems
- DOE Programs
  - DOE-NE spent fuel and waste science and technology program
  - Support hazard identification and characterization

Concluding Remarks

- Relying on a reasonable and flexible approach
- Sufficient capabilities to support non-LWR fuel cycle analyses
- Decades of model development and validation can be applied to non-LWR analyses as in Volume 3 and other programs
- Plan will be updated as more experience is gained and as new information becomes available