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UNITED STATES NUCLEAR REGULATORY COMMISSION'S
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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716TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

JUNE 5, 2024

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The Advisory Committee met via teleconference at 8:30 a.m., Walter L. Kirchner, Chair, presiding.

COMMITTEE MEMBERS:

WALTER L. KIRCHNER, Chair

GREGORY H. HALNON, Vice Chair

DAVID A. PETTI, Member-at-Large

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

VESNA B. DIMITRIJEVIC, Member

JOSE A. MARCH-LEUBA, Member

ROBERT MARTIN, Member

THOMAS ROBERTS, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

KENT HOWARD

P R O C E E D I N G S

(8:30 a.m.)

CHAIR KIRCHNER: Good morning. The meeting will now come to order. This is the first day of the 716th meeting of the Advisory Committee on Reactor Safeguards.

I am Walt Kirchner, Chair of the ACRS. Other members in attendance are Ron Ballinger, Vicki Bier, Vesna Dimitrijevic, Greg Halnon. Expect Jose March-Leuba to join us; Robert Martin, David Petti, Thomas Roberts, and Matt Sunseri. We also have our consultant Steve Schultz on the line virtually.

I know we have a quorum today. The committee is meeting in person and virtually.

The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act, FACA. The ACRS section of the U.S. NRC public website provides information about the history of this committee and documents such as our charter, by-laws, Federal Register Notices for meetings, letter reports, transcripts of full and subcommittee meetings, including all slides presented at the meetings.

The committee provides its advice on safety matters to the Commission through its publiclyavailable letter reports. Comments by individual members do not represent Committee decisions. The Commission speaks only through its published letter reports.

The Federal Register Notice announcing this meeting was published on May 10th, 2024. This announcement provided a meeting agenda, as well as instructions for interested parties to submit written documents or requests for opportunities to address the committee.

The Designated Federal Officer for today's meeting is Kent Howard.

A communications panel has been opened to allow members of the public to monitor the open portions of the meeting. The ACRS is inviting members of the public to use the MS Teams link to view slides and other discussion materials during these open sessions.

The MS Teams link information was placed on the agenda on the ACRS public website. Periodically the meeting will be open to accept comments from members of the public listening to our meeting.

Written comments may be forwarded to Mr. Kent Howard, today's Designated Federal Officer

A transcript of the presentation portions of the meeting is being kept. And it is requested that speakers identify themselves and speak with sufficient clarity and volume so they can be readily heard.

Additionally, participants and members of the public should mute themselves when not speaking, including cell phones, please.

During today's meeting the committee will consider the following topics: TerraPower Natrium Topical Reports on Principal Design Criteria, and Fuel and Control Assembly Qualification.

And we may get to commission meeting preparations.

At this time I'd like to ask other members if they have any opening remarks. Members? No?

I'm not hearing or seeing any.

And with that, I'm going to turn to Tom Roberts to lead us on in our first topic for today's meeting.

Tom.

MEMBER ROBERTS: Thank you, Chair Kirchner.

Good morning. Today we'll follow up on two nature and topical reports that were reviewed in the subcommittee meeting on May 15th.

I'll lead a discussion on the topical report for the principal design criteria. And my colleague Dave Petti will lead the discussion on the fuel and control assembly qualification topical report.

This was a pretty thorough review in subcommittee. Today we will hear a high level overview and then focus on residual questions from that meeting.

For the PDC topical report we'll focus on technical justification for the approach's plan for functional containment and the application of the SARRDL, or specified acceptable radionuclide release design limit concepts, since both of these seem to be major departures from past practice for Sodium Gas Reactors.

For the fuel qualification topical report we reviewed the major pieces of the qualification report and discussed how to support functional containment and the other safety functions of the design.

This morning's schedule allows for part of the meeting to be closed to protect TerraPower proprietary and export controlled information pursuant to 5 U.S.C. 552(b)(c)(4). If we need to do this, which I don't expect at this time but we'll find out depending on the discussion, we will close the public portion of the meeting and then restart the meeting.

I'll now turn it over to Candace de Messieres from the NRC staff for any opening comments she might have.

MS. DE MESSIERES: Good morning. And thank you for the opportunity to present today.

I am Candace de Messieres, chief of Technical Branch 2 in the Division of Advanced Reactors and Non-Power Production and Utilization Facilities in the Office of Nuclear Reactor Regulation, or NRR.

Today representatives from TerraPower and the NRC staff will continue discussions from the May 15th ACRS Kairos subcommittee meeting on TerraPower's principal design criteria, or PDC, and fuel and control assembly topical reports.

Both of these reports are used in reference in the construction permit application for the Natrium Reactor design for Kemmerer Power Station Unit 1 that was recently accepted for detailed technical review by the NRC staff on May 21st.

TerraPower's overall licensing approach

for the Natrium design follows the Licensing Modernization Project, or LMP, methodology. The Kemmerer Power Station Unit 1 construction permit application represents the first implementation of such an approach in licensing.

The PDC topical report describes the result of TerraPower's process to develop PDCs for Natrium using Regulatory Guide 1.232, Guidance for Developing Principal Design Criteria for Non-Light Water Reactors. The topical report was submitted in January 2023, was accepted for detailed technical review in March of 2023, and was the subject of an audit from September to October 2023.

The NRC staff's draft safety evaluation was issued on April 12th, 2024.

During today's presentation you will hear a summary of key design and regulatory features associated with TerraPower's PDC development approach, including context on the use of a functional containment and specified acceptable system radionuclide release design limits, or SARRDLs.

The fuel and control assembly qualification topical report provides TerraPower's plan to qualify fuel and control assemblies for the Natrium Reactor design. The topical report identifies acceptance criteria for fuel qualification and presents select fuel qualification results, in addition to ongoing and planned fuel qualification activities.

The topical report was submitted in January 2023, was accepted for detailed technical review in March of 2023, and was the subject of an audit from June through August 2023.

The NRC staff's draft safety evaluation was issued on March 20th, 2024.

Thank you again for your time and consideration. And we look forward to the discussion today.

MEMBER ROBERTS: Thank you.

George, are you going to start it?

MR. WILSON: We greatly appreciate -- I'm George Wilson, Vice President, TerraPower. We greatly appreciate the time of the ACRS to present on our two topical reports for Fuel Qualification and Principal Design Criteria.

And with that, I'll turn it over to Ian Gifford.

MR. GIFFORD: Thank you very much.

My name is Ian Gifford. I'm a licensing manager on the Natrium Project. We'll start today's discussion with the fuel and control assembly qualification presentation by Dr. James Vollmer. He is participating remotely.

James, are you able to hear us?

MR. VOLLMER: Yes. I hear you fine.

Ready for me to start?

MR. GIFFORD: Yes, please.

MR. VOLLMER: I think most of you already saw this, so I'll go fairly quickly.

MEMBER ROBERTS: Just to clarify, it's fine, our intent was to have TerraPower present and then the staff respond. If you care us to go in a different order and present the fuel qualification first, just -- we'll cover both before we turn it over to staff. Is that right?

Okay, thank.

MR. VOLLMER: So, I'm James Vollmer from TerraPower. I'll provide a quick overview of the fuel and control assembly qualification topical report. I'll go fairly quickly since I think most of you have seen this before. But feel free to slow me down or stop me if I'm going too fast.

Next slide, please.

So, this is a brief high level overview of the Natrium Reactor. Some key features we want to call out.

That we are using metallic fuel that has been used historically, especially within the DOE program, for Sodium Fast Reactors.

Has very high compatibility between the metallic fuel and sodium. Good retention properties of the metallic fuel matrix to retain fission products within the matrix itself.

Good compatibility between the two. If there were to be a breach, large thermal inertia for the large sodium pool within the reactor itself to help promote cooling of the reactor with natural convection.

And then we also have an additional air cooling passive system for the old reactor vessel to help maintain coolability of the reactor under all conditions, accident scenarios.

Likewise, for the control assemblies they are gravity-driven. But then we also have a motor-driven control rod runback and scram follow feature as well.

And just inherently stable core with increased power or temperature.

We will rely heavily on our program from the historic U.S. SFR experience, EBR-I, EBR-II, FFTF.

And then we will use, rely on the TREAT tests that were done historically, as well as perform the tests in this reactor.

Next slide, please.

This is a brief overview on our approach. So, actually, we started our fuel qualification efforts engagement with the NRC in 2019 through a DOE grant, regulatory assistance grant.

As part of that, we were relying heavily on NUREG-0800. And interpreted how it applies to Sodium Fast Reactors with metallic fuel. Adapted the requirements specifically to -- well, we're directly applicable to metallic fuel, which are not some that were not identified that were, we though, were needed for metallic fuel systems and Sodium Fast Reactors. So, we call these Regulatory Acceptance Criteria.

We did submit three White Papers and received feedback from the NRC as part of this process.

Next slide, please.

So, given the large amount of pre-engagement we already had and have our test programs aligned with this, that was the overall structure we used with the topical report since NUREG-2246 actually came out fairly late in our process. But we did include a section directly mapping between our approach and NUREG-2246 to show that we meet all the assessment framework goals identified, except for the one that was not directly addressed was G2.2.1, radionuclide retention requirements.

And that was specifically addressed by separate submittal from TerraPower's Radiological Source Term Methodology Report.

Next slide, please.

So, just a high level overview of the methodology.

So, as I mentioned, we identified the regulatory acceptance criteria. For each one of those acceptance criteria we made sure we have a design criteria and the basis for that.

And then we included a fuel system description to make sure we can define the fuel system in enough detail that a regulatory can understand the overall design. And that's the basis of our analyses.

The design evaluation includes historic operating experience, testing, as well as methods.

And then we also included brief sections on testing and inspection of the fuel as well as an ongoing surveillance program within the reactor plant.

Next slide, please.

A key aspect of it is we did perform PIRT analysis to identify for each individual design criteria. What were the key phenomena that we needed to understand well to ensure that we met the associated limits?

So, here are some examples:

Thermal creep strain in the cladding is an actual failure criteria that we used. Its purpose is to prevent cladding rupture or coolant flow blockage.

And then kind of the key phenomena that influence that criteria:

So, the HT9 cladding properties in this particular model. Fuel-cladding chemical interaction, cladding wastage since that thins the cladding wall, and then the fission gas release within the fuel itself because the more the fission gas retained within the fuel matrix and the strain or the stress on the cladding is higher for even more strain.

And, likewise, we have the total strain limit that includes the impacts of irradiation and creep swelling on the cladding itself. And, again, that's mainly to preserve coolant channels between the fuel pins.

We also have fuel temperature peak fuel cladding or peak cladding temperature, and then an overall cladding wastage criteria.

Next slide, please.

So, here's just an overview of our fuel design.

So, the center image is the Type 1 fuel pin cross section. So, the green represents the U 10 weight percent zirconium that has been tested extensively within the DOE program.

The yellow represents the sodium bond between the fuel and the cladding. That's intentional, to provide space between the fuel and the cladding so the fuel during irradiation can swell outward and get interconnected porosity that promotes release of the fission gas up to the fuel plenum. So, the sodium is simply there to conduct the heat out until the fuel expands outward to touch the cladding.

On the right you see an axial cross section of the fuel pin. So, you have an axial shield slug below the fuel slug. And then the sodium bond actually comes up above the fuel column at the beginning of life. And the fuel expands radially and axially with irradiation. And then the sodium will start backfilling within the fuel once that porosity interconnects.

MEMBER PETTI: James.

MR. VOLLMER: Yes?

MEMBER PETTI: Just a question.

That cross section labeled Type 1, that's not to scale, is it?

MR. VOLLMER: Not for actual dimensions.

MEMBER PETTI: Right.

MR. VOLLMER: But to relative scale it is, yes.

MEMBER PETTI: Oh. So, like, the yellow is the sodium is as thick as the cladding?

MR. VOLLMER: At the beginning of life, yes.

MEMBER PETTI: It is. Okay, good. Thank you.

MR. VOLLMER: And then here's the hexagonal --

CHAIR KIRCHNER: May I follow up, James? This is Walt Kirchner.

MR. VOLLMER: Yep. Go ahead.

CHAIR KIRCHNER: Yeah. So, just give us a feeling for the performance. When do you expect nominally the fuel expansion to displace the sodium and make contact and then --

MR. VOLLMER: Typically -- Go ahead, sir.

CHAIR KIRCHNER: And then just explain a

little bit further about how the expansion is accommodated as it goes from radial to axial. Could you just talk through the fuel on this?

MR. VOLLMER: Yep.

Yeah, so, roughly kind of 2 percent burnup is kind of typical where the fuel contacts the cladding. So, that would be within our first irradiation cycle in the reactor. The fuel would expand radially and make contact.

And, again, it also expands axially not exactly at the same ratio but close to the same ratio. And as soon as the fuel makes clad -- contact with the cladding its axial expansion slows down and basically stops at that point. So, limited axial expansion past that initial growth point.

And it is largely driven to irradiation growth within the metal, which itself involves fission gas -- excuse me, fission gas pressurizing the fuel. And it really is kind of opposite of Light-Water Reactor fuels where the fuel is very soft, the cladding is very hard so the fuel does behave much more like a putty almost, if you will, to some extent.

Does that address your question?

CHAIR KIRCHNER: Once you make contact then the, then the further expansion is taking up axially?

MR. VOLLMER: It is. But it's actually pretty limited because by that point you have a full connect network of porosity within the fuel so that the gas is released to the plenum at that point in time. So, your driving force to expand is reduced. So, it doesn't keep growing axially typically beyond that point.

CHAIR KIRCHNER: Thank you.

MR. VOLLMER: Yep.

Next slide.

So, these images, again, are attempted to be to scale to each other. So, it kind of shows the EBR-II cross section of the fuel pin. The larger metallic fuel, the MFF fuel assemblies, and FFTF, and then the Natrium Type 1 fuel.

And, so you do see the Type 1 is slightly larger than the MFF fuel but it is within what has been tested historically in other metallic fuel test designs.

You also see here's a cross section on the right side of height between the different fuel assemblies. You do see that the FFTF fuel column is much taller than the EBR-II. And although the Natrium Type 1 overall fueling fuel column, fuel -- sorry, fuel assembly link is much taller, the fuel height is actually almost exactly the same as the FFTF fuel column. And that was intentional that we did not want to extrapolate beyond what was tested for the overall fuel height.

Next slide, please.

So, from a fuel system design evaluation, we are relying heavily on the historic operating experience because there are no longer any Fast-Operating Reactors in the U.S., plus there is a wealth of historic data that we were able to rely on. So, we have been working for many years with the DOE labs to obtain legacy fuel data as well as qualify it.

Not only does this include full fuel pin irradiation tests but also has fuel, and material properties and transient and accident tests that we were able to use.

We do have several other test activities in progress or planned, including, so the FFTF fuels are most reflective of our fuel design. Most of those were not looked at after irradiation. So, we have sponsored additional post-irradiation exams to address some of the gaps in the historic database and just demonstrate its performance relative to the EBR-II fuels to get a fuller picture and understanding.

Also, transient testing of fuel pins. So, the TREAT reactor in Idaho was specifically designed for severe accident testing of fuels to test to failure. That has been re-started recently, so we do plan on testing full length irradiated metallic fuel pins from the FFTF reactor, and then also additional furnace tests to just better characterize the transient behavior in severe accidents.

We have fuel and absorber property tests, including we created metallic SIMFUEL, so we simulate burnup in the fuels by adding representative fission product species to it.

We have a host of HT9 materials tests.

And then core assembly and mechanical tests as well.

Next slide, please.

So, for our materials test programs we actually kicked these off 2011 time frame. Our first step was actually to get HT9 materials. So, we actually worked with multiple suppliers and got three unique heats of HT9 just to characterize the bounds of the specification, as well as understanding the impacts to performance.

We actually chose those compositions based on the historic operating experience of the DOE materials and looking at those materials.

Some of the key gaps we felt from the historic database was expanding the time at temperature for those. We did an extensive thermal aging program up to 50,000 hours just to understand are there any microstructural changes in it just due to time at temperature?

And then doing microstructural characterization and mechanical testing on those.

Again, we did actually make new heats of HT9 material just because we wanted to verify how it performed relative to historic HT9s. We have side-by-side irradiations of that material with some of the legacy DOE material in the BOR-60 reactor in Russia, and have it irradiated up to about 85 dpa for that.

We also have planned irradiation tests on welds and coatings, and some advanced materials. This is on the High FIR Reactor in Oak Ridge National Lab.

And then thermal creep testing is another kind of long time at temperature phenomena we were concerned about. So, we have tested up to 70,000 hours for thermal creep testing for HT9. And then also have axial tube creep underway and biaxial tube.

And really the purpose of these tests is to help us refine our overall response models for the

HT9 behavior.

MEMBER PETTI: James, in terms of the radiations in BOR-60.

MR. VOLLMER: Yep.

MEMBER PETTI: They're complete? Are the samples back in the States yet?

MR. VOLLMER: Not yet. We are close to shipping them back. I think they're in the process of packing and queuing them up right now.

MEMBER PETTI: Okay. That's good.

I remember when it started in the DOE program.

MR. VOLLMER: Yeah. Yeah, it's been a bit of a journey to get them back.

MEMBER PETTI: A long time coming.

MR. VOLLMER: Yes, yes. But we're close.

MEMBER PETTI: Good.

MR. VOLLMER: Next slide.

So, this is just a brief overview of some of our fuel performance tools at TerraPower. So, from a fuel pin performance point of view.

So, we have two codes crucible. It's a fast-running code that's actually integrated with our overall core design software, the ARMI software. And it really is aimed at the key phenomena that are tightly coupled with neutronic responses for, like, the fission gas release, the sodium-bond, infiltration, fuel axial growth.

But we also do have some of the fuel performance phenomena like cladding wasting -- cladding wastage, clad temperatures, cladding strain. So, as they are iterating the core, they can verify the ARMI where we expect to meet our fuel's design criteria as part of that process.

But once they have the overall core designs that they think meets all those goals, then they will give us individual assembly fuel pin histories so that we can perform our detail analysis with our ALCHEMY Package, which is a finite element base method.

On the right is an example of a cladding tube where you can see the different finite elements and the actual predicted strains along that fuel pin.

Again, it is a high-fidelity model. Captures all the phenomena we think are key for modeling metallic fuel behavior, fission gas release, FCCI, thermal conductivity. But then, in addition to fuel, we are able to adapt models for the boron carbide as well, so we can use the same tool for our absorber predictions as well as our fuel predictions.

And then we also have models specifically designed to support our ongoing irradiation tests so we can pre-test predictions ahead of time, run the actual tests, and then analyze the results and compare. So, it helps validate the model in the process as well.

Also, the ALCHEMY Package is used as kind of the structural material models for the ATR material that are used at our higher linked scales, in particular OXBOW for our full fuel assembly models and for restraint system. And the materials come from ALCHEMY for those.

Next slide, please.

So, I mentioned OXBOW. That's our primary core mechanical performance tool. Can do single assembly just to verify kind of the amount of distortion anticipated within a fuel for core assembly. And then prediction, kind of withdrawal and insertion tech loads based on those distortions. But then also from a core-wide, core lockup response as a function of thermal or irradiation behavior, can use the same tool.

And then also perform it for seismic analysis as well.

We also have a module within OXBOW that

also do control assembly scram under seismic tech response and assembly drop times, control assembly drop time. And then bundle-duct interactions as well.

Next slide, please.

We have extensive testing underway for our core assembly response. On the right side this is actually a fixture we use for the mechanical testing. We can actually fit multiple fuel assemblies within that.

In the center picture, that's actually within our Bellevue Lab of a pit with that inside of it where full length fuel assemblies can be distorted, put in there, measure the withdrawal insertion loads to pull them in and out.

Likewise, we can load multiple assemblies, apply thermal gradients, verify the bending response of them.

On the bottom right shows a sample of kind of a bundle compression test just to look at how does the bundle redistribute with loads applied from given bases of the assembly.

Have a whole host of kind of single assembly, multiple assembly, and then these bundle-duct type interaction tests as well.

And then we also have worked with the

international community to benchmark against historic codes as well as historic databases for operating Fast Reactors.

MEMBER PETTI: Then, James, just a question on that.

MR. VOLLMER: Yes.

MEMBER PETTI: The rest of the world is in oxide space.

MR. VOLLMER: Yes.

MEMBER PETTI: Is it still valuable? I mean, you've got then the metal system vs. the oxide system?

MR. VOLLMER: Yeah, very much so, that the thermal gradients within the fuel assemblies are largely the same and have the exact same radiation effects. And I did see DOE HT9 material, so it was actually an oxide fuel assembly achieved the highest DPA on. So, we've been using that assembly extensively for benchmarking for the dilation and whatnot due to radiation performance.

So, it's very relevant overall.

MEMBER PETTI: Okay. Thanks.

MR. VOLLMER: Yeah.

Next slide, please.

Kind of really the last piece of our

qualification program is our fuel surveillance program.

So, we did actually design Type 1 fuel to be very conservative relative to its historic designs. And believe there is enough margin in lifetime we can connect it to an additional cycle. But we've restrained it to what the historic operating experience was just to verify we are bound by history.

But we do have special fuel assemblies, we call them our Lead Demonstration Assemblies, that do have pins that we can remove in these, the X fuel handling to pull them out for expedite post-irradiation exams. So, that way we can constantly monitor performance throughout lives.

So, after each cycle be able to pull pins out, do visual exams on them, measure them, send them off for extensive post-irradiation exams or structural exams just to make sure that the fuel is behaving as predicted based on the historic operating experience.

We will be targeting a subset of them to actually have accelerated burnup so we'll actually maximize the enrichment of those pins so that they are dating the rest of the core as far as burnup.

And then also trying to target bounding conditions for some of them as well just to verify that we are bounding the entire performance of the reactor.

Next slide, please.

Yeah, I think that's it for me. So, any questions?

MEMBER ROBERTS: Yes, a question.

If you can go back to slide 6. This slide is a, it's how the PIRT Evaluation Identifies Fuel Phenomena.

MR. VOLLMER: Yes.

MEMBER ROBERTS: And it appears to be a list of design limits --

MR. VOLLMER: Yes.

MEMBER ROBERTS: -- on the fuel.

MR. VOLLMER: Yes.

MEMBER ROBERTS: Such that when you do the analysis, either safety analysis or steady state, you would go verify those five limits are met. Is that right?

MR. VOLLMER: Correct. Correct.

MEMBER ROBERTS: So, that sounds like you would transition to the PD2. It sounds like a SAFDL.

MR. VOLLMER: That's correct.

MEMBER ROBERTS: Just wanted to understand what the difference was. If you intended to meet all these five limits and then call that a SARRDL, I'm just trying to understand what the, what the overall intent is.

Because it sounds like your intent is for the design to meet these five limits.

MR. VOLLMER: That's we do design to them and then what happens then goes off to SARRDL space. So, that's where we have our damage criteria. That would be basically to say that's when the fuel's reached its effective lifetime. So, if you go through an AOO, and then beyond that we can see the fuel damage, you would not reuse the fuel past that point.

But then we do have failure criteria. So, if you did exceed that, we would say the fuel has been failed. And then that would go over to SARRDL space for them to propagate what's the impact of that failure.

MEMBER ROBERTS: So, for normal operation AOOs, which is what the JDC or PDC states, you would expect to have a zero release by having met these five criteria? Is that what would be here?

MR. VOLLMER: Exactly.

MEMBER ROBERTS: Okay.

MR. VOLLMER: Exactly, yep.

MEMBER ROBERTS: And then for more severe events you would simply calculate the amount of damage. That's not a SARRDL, right, that's a design basis calculation of what the consequence is; is that right?

MR. VOLLMER: So, I think both. We basically calculate how many fuel pins are potentially failed. And then it would go on the SARRDL space to understand what would be the dose potentially released from that.

MEMBER ROBERTS: Right. Release in a containment and then the whole, you know, down in those type conditions.

Okay, thanks.

MR. VOLLMER: Any others?

MEMBER PETTI: So, just to make sure I'm clear. Then you really have both SAFDLs and SARRDLs, depending on the space of the, of the accident domain, if you will.

MR. WILSON: This is George Wilson from TerraPower.

We'll discuss this more in the PDCs. So, if you'll wait till we get to the PDCs --

MEMBER PETTI: Perfect.

MR. WILSON: -- and ask additional questions we'll go into a little more detail.

MEMBER PETTI: I have another. I have a question on the surveillance stuff, which I think was really good.

How often do you anticipate pulling the pin out of that test assembly?

MR. VOLLMER: For the initial cycle --

MEMBER PETTI: Right.

MR. VOLLMER: -- through the entire fuel lifetime, every cycle we'll be pulling a subset of pins out. And I guess just to clarify, we're not putting the fuel assembly back in after we pull it out.

MEMBER PETTI: Back in, right. I figured that, not replacing it.

MR. VOLLMER: Yes.

MEMBER PETTI: But do you have to put something back in the core, though? Just, or there would just be a hole?

MR. VOLLMER: Well, just replace it with a fresh fuel assembly.

MEMBER PETTI: Oh, you put it back. Right. Got you.

Okay. Thanks.

MR. VOLLMER: Yep.

CHAIR KIRCHNER: Jim, just on that topic.

Will those be wire wrapped or they'll be kind of a straight pin? How do you compensate for the non-prototypicality that you get out of those?

Is there any issues that you are seeing of not, not having that pin that you removed wire wrapped?

MR. VOLLMER: They will still have the neighbors will have wire wraps. So, they will still have the support. It will be a small perturbation. And we are doing thermal hydraulics testing just to verify what it is. But we do anticipate it would be a fairly small difference on it.

If anything, they'll likely have a little more propensity to move. So, a little more spreading type interaction.

CHAIR KIRCHNER: Right.

MR. VOLLMER: But, again, we expect it to be very small, just that the bundle is so tight that there really is not much room for movement.

Yeah, we're doing harmonic testing to verify that.

CHAIR KIRCHNER: Okay, thank you.

MR. VOLLMER: Yep.

MEMBER PETTI: Just another question comes to mind.

What irradiation testing of the welds, --

MR. VOLLMER: Yes.

MEMBER PETTI: -- coatings, and like, you're not going to get the DPA. Is there some historic data so that you can kind of make correlations based on, you know, this stuff looks as good as the old stuff, so we can use the old, the old stuff?

MR. VOLLMER: Yeah. We do. There's a lot of kind of TIG welding I think historically was primarily used. We are wanting to use a different welding process. But it should have a smaller heat-affected zone and whatnot.

But also, the welds typically are out of the high flux area.

MEMBER PETTI: True. Yeah, right.

MR. VOLLMER: They don't receive much of the dose as well.

MEMBER PETTI: What sort of dose are you going to get?

MR. VOLLMER: For the welds?

MEMBER PETTI: Yes.

MR. VOLLMER: For most of them it will be, I think, less than 5 gpa, as I recall.

MEMBER PETTI: That's my guess. That is

what I guessed. Okay.

MR. VOLLMER: The control assemblies, that is the one where it will actually be in the higher flux region of the core. So, that was kind of the area we want to make sure we do a bound aspect of it.

CHAIR KIRCHNER: Jim, what do you think -- this is Walt again -- what do you think of your list of parameters you're testing for, what do you feel is the kind of the limit for your, you know, your fuel design for this, for this application?

MR. VOLLMER: So, FCCI is kind of the most, like, limiting in our experience, continues kind of the hot channel factors. So, that's why we spent the most effort for our post-irradiation exams we are doing on the FFTF pin, but specifically looking at the FCCI response which we think that is, again, likely to limit.

Because the DOE fuels were high enriched, so they would be higher linear power since they're lifetime faster. So, kind of the time at temperature combination. So, that's really what we want to make sure we understand that phenomenon well.

CHAIR KIRCHNER: Thank you.

MR. VOLLMER: Yep.

MEMBER PETTI: Just one more since you've got such a wealth of knowledge.

All of the out of trial thermist tests that I remember reading the report that gives you the rate of attack, if you will, in that plot I can remember, does the in-pile stuff agree generally well with the out-of-pile there, I mean given all the uncertainties?

MR. VOLLMER: For the FCCI, the historic models found no effect of irradiation. We've actually for our SIMFUEL stuff we found the out-of-pile to be more aggressive than the in-pile, but we believe that's just because we haven't been able to recreate the mixture of the fuel products quite like was the actual irradiated fuel itself.

MEMBER PETTI: So, it's going to serve it?

MR. VOLLMER: Right.

MEMBER PETTI: I mean, there are examples in LWR space where they tried to make fuel, simulated fuel, and it was found to be grossly over-conservative than --

MR. VOLLMER: Yes.

MEMBER PETTI: -- what you see in-pile. But, you know, that took two years to get there. And look back and that was a really dumb idea, you know. You understand why we do it, because it's easier. But you run that risk.

Okay, good.

MR. VOLLMER: Yeah, we're relying strictly on the in-pile data for our FCCI correlation. The out-of-pile is more to try to understand mechanistically if we can refine the model, or just to give us more insight into how to model the data.

So, yeah, we kind of gave up on using it from a purely mechanistic under pure correlation. Just want to do more just a qualitative insight of the behavior from our out-of-pile tests.

It's time for me to turn it over to Ian.

MR. GIFFORD: Thank you, Jim.

So, I'll provide a brief overview of the methodology that was used to develop the principal design criteria for the Natrium Advanced Reactor. And then I'll turn it over to Eric Williams, who is our senior vice president and design authority, for a focused discussion on SARRDL and functional containment.

The approach to PDC development was discussed with NRC staff during public meetings in December of 2021, and November of 2022. And the PDC topical report was submitted in January of 2023.

In accordance with 10 C.F.R. 50.34,

Principal Design Criteria were also included in the construction permit application, Section 5.3.

Regulatory Guide 1.332 provides guidance for Non-Light Water Reactors to develop principal design criteria for Non-Light Water Reactor design. The Reg Guide acknowledges that different requirements may need to be adapted for Non-Light Water Reactor designs, and that the PDC in 10 C.F.R. Part 50, Appendix A, are you regulatory requirements for Non-Light Water Reactor designs, but they provide guidance in establishing the PDC for Non-Light Water Reactor designs.

Ultimately, it's the responsibility of the applicant to development PDC for its facility based on the specifics of its unique design.

Applicants are allowed to use the Reg Guide to develop all or part of the principal design criteria, and are free to choose amongst the Advanced Reactor design criteria, Sodium-Cooled Fast Reactor design criteria, or Modular High Temperature Gas Reactor design criteria to develop each piece.

PDC were developed starting with the SFRDC. And in Appendix D of Reg Guide 1.232, first discussion was whether the PDC applied. And if it did it was assessed for whether it could be adopted as written.

If it could be adopted as written, it was accepted as an initial Natrium PDC.

If it was needed to be modified, we first reviewed the ARDCs and the MHARDCs for language that may be more applicable to our design. And we also left open the option that we may in fact have to draft new PDCs for Natrium.

I want to focus a little bit on the box here. So, we have the initial Natrium PDC list and then the box that says "perform the iterative LMP process."

So, the LMP is an iterative process throughout the design phase. NEI 18-04, states that Reg Guide 1.232 should be used as an input by designers to initially establish principal design criteria for the facility based on the specifics of the design

And then, as part of the LMP process, PRA safety functions are identified that are necessary and sufficient to meet the frequency consonance target for all design basis events and high consequence beyond design basis events to conservatively ensure that 10 C.F.R. 50.34 dose requirements can be met.

The PRA safety func -- these PRA safety

functions are then defined as required safety functions, or RSFs. RSFs are used to develop required functional design criteria, RFDCs, that establish reactor-specific functional criteria that are necessary and sufficient to meet the required safety functions.

NEI 18-04 states that the required functional design criteria, RFDCs, are defined to capture design-specific criteria that may be used to supplement or modify the applicable GDCs or ARDC in the formulation of principal design criteria.

The Natrium Project has undergone a complete iteration of LMP to include a thorough review by the Integrated Decision Making Process Panel. In accordance with NEI 18-04, the Natrium LMP Design Criteria Report includes a complete mapping of LMP evaluated functions.

RFDCs developed from the LMP are all mapped to at least one principal design criteria, demonstrates that the PDCs are complete with the current Natrium design. No RFDCs were found that would require a new or expanded principal design criteria.

As we progress the design and continue with LMP, the completeness of the PDC will continually be revisited. Significant changes to the principal design criteria are not expected, but should they be needed it would be appropriately communicated to the NRC.

MEMBER MARCH-LEUBA: This is Jose.

Let me emphasize what you just said I'm 100 percent I'm in agreement with. But when during that slide it says we are starting with a set of design concepts, and then we remove the ones that don't apply.

We need to see is there something special with my system that requires a new one. The thing that comes to mind is now they're available to us through the source.

So, I'd like it that you're seeking that one.

MR. GIFFORD: Appreciate the comment. Thank you.

MEMBER PETTI: So, then it's fair to say that the PDC Report that we're reviewing represents the final PDCs in the bottom box or the initial PDCs? Or are they the same based on where you are?

MR. GIFFORD: They are the same based on where we are.

MEMBER PETTI: Great.

MEMBER ROBERTS: I have a similar question, is the topical report doesn't have those bottom two boxes. And there is a limitation in condition in the safety evaluation that does tie to that a little bit. It says that the NRC acceptance is based on an understanding you are using LMP process, so that pretty well drives what you just said.

Is there an intent to change the topical report or is the reference from the SCR considered to be sufficient to ensure that a future user does what you just said and not what the topical report says?

Let me pull this up. It has this figure without those bottom two boxes. And there's no discussion, at least that I recall, in the report itself of that LMP iteration.

I believe I think it's very important. And it sounds that you do, too. So, it's just a matter of making sure whoever uses the topical report understands what you just said.

MR. GIFFORD: Yes. The intention would be whoever uses this topical report would be using NEI 18-04. And so, following 18-04 would require that an applicant would go through those iterative steps.

I think the intention of the figure in the topical report is to show how the PDC were developed at the time that the topical report was submitted in January 2023. And then we have subsequently followed the LMP process and moved that into the construction permit application.

MEMBER ROBERTS: Okay, thanks.

MR. GIFFORD: At this time I will turn it over to Eric Williams for a discussion on SARRDLs and functional containment.

MR. WILLIAMS: All right, thank you.

So, my name is Eric Williams, Senior Vice President and Design Authority, TerraPower.

I was, you know, reflecting on the questions that were asked in the last meeting and trying to, you know, come up with the best approach to try and get some clarity behind these issues. And, you know, coming at it from the design perspective I first just wanted to mention a couple of the new things that TerraPower is doing in the design of Natrium I think are incorporating a lot of this new material on SARRDLs and functional containment.

So, you know, using the LMP approach is the first thing, you know, that's sort of new here, following the risk-informed performance-based approach. And then the use of something like a SARRDL becomes really integrated closely with that approach.

And so, talking about them in separate conversations becomes really hard.

We're also following a systems engineering approach to design. So that means that we're trying to rigorously set functional design requirements at the beginning, including safety requirements that all of these things factor into so that the designers can have clear requirements as they go through their iterations and know that they're meeting an acceptable design.

And then, finally, we're using the IAEA framework that calls out defense line functions where it allows us to look at design requirements in each of those defense lines, so that as the designer is moving through their work they can also be evaluating defense-in-depth adequacy as well.

And then, of course, through the Integrated Decision Making Process Panel that's part of LMP, we get a chance as a group to review that defense-in-depth and how we're meeting the frequency consequence limits with margin in the design.

So, what you get out of that is a really great integrated set of design and safety requirements. But it also means that these things are hard to pull apart from each other.

So, that's kind of the framework that we're applying. And so, I'll start by talking about SARRDLs first and then that will roll into functional containment. And they're really closely connected.

So, we've already talked a little bit about SARRDLs.

So, what is a SARRDL? First thing we do is we look at all the systems and components that are containing radionuclide inventory in the plant. Sometimes storing it, sometimes circulating it during operation. And we're trying to set clear limits on potential releases for that circulating radionuclide inventory during normal operations or AOOs.

So, we look at each one of those radionuclide-containing systems. And if it has the potential to violate a release limit, then it gets a SARRDL. And that SARRDL is usually in the form of a volumetric leakage rate that can happen from that system.

And so what it allows us to do is set a clear design requirement on those SSCs that get the SARRDLs so the designers can use those in design and know that they're meeting the requirements.

It's also very convenient to establish design -- or analysis assumptions. So, when you're looking for the worst case initial condition to initiate, say, a DBA analysis, you can initiate it assuming that you're at the SARRDL limit in that system.

So, that's a convenient way. And it makes me think back to Light Water Reactors that may assume you're at your worst case tech spec limit of primary system activity when you initiate an accident. So, that's a really convenient way to do it.

And then, in the end those SARRDLs get incorporated into the frequency consequence curves that you see with LMP that ultimately show with specific PRA datapoints showing up on the F-C curve with uncertainties identified that shows that you have margin to the F-C curve limits.

And that ultimately is a practical way to demonstrate the PDC 10 compliance where the SARRDLs are mentioned.

So, that's how the SARRDLs are used. And, you know, the SECY paper 18-0096 talks about how those are closely linked with functional containment because they express the limits at a performance criteria for functional containment for the AOOs.

So, we can talk a little bit more about functional containment now. And I wanted to kind of start with a couple of the design features because I know one of the questions was really talking about the rationale --

MEMBER ROBERTS: Excuse me, Eric.

MR. WILLIAMS: Oh, sure.

MEMBER ROBERTS: GDC 10, just to understand how that works with the SARRDL concept.

What we heard in the previous presentation is the fuel limits are going to be tracked as design limits. So, essentially you have no release from fuel during normal operation or AOOs. But then you said that the SARRDLs become other circulating activity and other radiation-containing systems, --

MR. WILLIAMS: Yes.

MEMBER ROBERTS: -- not the fuel.

MR. WILLIAMS: Uh-huh.

MEMBER ROBERTS: That seems like a different interpretation of that PDC -- or GDC. The purpose of that GDC seems to be that you not have any challenge to radionuclide release from AOOs or normal operation by keeping all the circulating activity within the fuel.

Sounds like you're doing exactly that. But then you've added, because of, I guess, the LMP and the need to have a downwind dose reduced in normal operations and AOOs, expanding what GDC 10 is, PDC 10 currently says to include circulating activity. Is that the right interpretation?

MR. WILLIAMS: Let me try and think about that.

MEMBER ROBERTS: Because it seems like the intent of that PDC is to not have release from fuel during normal operation or AOOs.

MR. WILLIAMS: It's really to not have any releases that violate the 10 C.F.R. 20 limits that are imposed on normal operation and AOOs by establishing clear requirements for all the radionuclide-containing systems.

If we meet the fuel design limits perfectly, then there won't be any from the primary system.

There can also be, you know, radionuclide inventories within systems like sodium processing systems, sodium cover gas from some prior failed fuel that occurred, you know, even just randomly. So, you have to look at those.

You have refueling systems that have to be looked at, too, because they contain fuel at times.

And so, the SARRDLs look at all of those systems and incorporate these limits, not just in the primary system.

But we do use the design limits, as James was pointing out. We look at those in terms of the mechanistic source term methodology. So, it would determine the failed fuel fractions that would get incorporated into mechanistic source term.

But the real intent of the SARRDL is to capture what is really the phenomena that is really important for a Sodium Fast Reactor. Since there's not a direct, as direct coupling from fuel fraction, fuel failure fraction to radionuclide release, because we have all these extra systems that act to, you know, attenuate radionuclide release, we have to incorporate all of that. And the SARRDL does that.

And so, it's really a better metric of what is happening in a Sodium Fast Reactor. And that's kind of the intent of this is to address it directly on what's happening.

But we're still using the fuel design limits, like you said, as part of the mechanistic source term process.

MEMBER MARTIN: Eric, this is Member Martin.

You know, what you describe doesn't sound any really different than what we've always done.

It's always been a kind of analytical defense-in-depth. You know, the ultimate metric is the SARRDL; right? We're interested in doses and its comparison to 10 C.F.R. 100.

That's backstop. In fact we can demonstrate the SAFDLs. We have high confidence on dose. And then those SAFDLs are backed up by tests. We've always done it that way.

I guess I'm not seeing something new here, except for maybe a documentation emphasis in SARRDLs. And maybe, and that's consistent with LMP, no doubt. But in practice, which I think makes LMP practical, is that it's not a big departure from what we do. Obviously, LMP brings in a lot of the risk aspects.

And so, it's another way of defending any kind of engineering judgment, you know.

MR. WILLIAMS: Yeah.

MEMBER MARTIN: But I'm not really seeing anything new here.

MR. WILLIAMS: Yeah.

I don't think of it as that new, other it's more integrated because in the LMP approach you're going to see all of the results together in this frequency consequence curve. And the SARRDLs are consistent with the way DBAs are also looked at, and

DBEs are looked at.

So, it all forms this, like, self-consistent way of talking about the narrative of safety and showing the margin that we have in design safety. And it retains a significant amount of margin, too, and it shows you where that margin is. Right?

It's not new but it's a new way of talking about it. Hence, the reason for this.

MEMBER MARTIN: All right. It sounds like the key is the predicate of the SARRDL is to meet the fuel limits. And so, that gets you basically unchanged from existing Sodium Fast Reactors or even Light-Water Reactors. And by whatever technology, high temperature gas we had to do it a little different.

MEMBER PETTI: Yeah. I think that's in my mind the difference because there are no SARRDLs because it's difficult in that system to take because it's not a clad, pin and clad system.

So, basically, you know, even though the SARRDL is there and it's to demonstrate margin against top 20, there's even more margin when one looks at the design limits that you have that you're going to say zero.

MR. WILLIAMS: Uh-huh.

MEMBER PETTI: Or some very low number

But we're going to analyze it up here with SARRDLs and it will be higher, there's even more numbers.

MR. WILLIAMS: Right. Exactly.

MEMBER PETTI: And once you get beyond AOO space and it's a, it's a calculation based on fuel performance leading to the larger source term, if you will.

MR. WILLIAMS: Right.

MEMBER PETTI: Okay.

MR. WILLIAMS: Yep.

MEMBER ROBERTS: Is that different than other reactor types? That sounds like the same thing with Light-Water Reactors as when you get into, say, LOCA space you've got to go calculate what their relief fractions are or bound it conservatively.

It sounds are you doing anything different there?

MR. WILLIAMS: I think, I think the difference is in there you're going to have a lot of additional features coming into play through the mechanistic source term analysis than a traditional Light-Water Reactor would have. I think more recent

Light-Water Reactor applications have looked at, you know, fission product deposition and containment, crediting additional things like that.

We're doing something like that, plus a lot more because of the Sodium Fast Reactor features and the lack of the pressurized system.

MEMBER PETTI: I think if you went back in, you know, in the olden days, you know, with the TID source term, that was just sort of an analysis. But the 1.183 that we just looked at really is a culmination of LWR source term that has more mechanistic stuff behind it.

MR. WILLIAMS: Right.

MEMBER PETTI: But the reactivity events do this, the LOCAs do this, and it was a way to capture all of that. So, you're basically kind of doing the same thing with the SARRDLs, the technology is the same.

MR. WILLIAMS: Yeah, yeah.

MEMBER ROBERTS: Right. More mechanistic. And just taking the worst of all accidents and finding them.

MEMBER PETTI: Right.

MEMBER ROBERTS: But, fundamentally it's, it's you go run your analysis, figure out what the release is because you violated the fuel limits because of the nature of the accident. And then you figure out what that is and you proceed with a calculation.

MR. WILLIAMS: Yeah. We are still demonstrating that we need the functional containment performance criteria with an assumed major accident. So, we are still doing that even though it's probably in the -- beyond the cutoff frequency in the PRA.

So, we are still taking that step. But we are crediting all of the design features that we have that are technology specific.

MEMBER ROBERTS: Sure. And that's consistent with Sodium Fast Reactors in the past; right? They looked at the protected loss of flow, and your protected transient input powers, and those are probably extremely low in frequency space.

MR. WILLIAMS: Right.

MEMBER ROBERTS: I'm sorry. I guess you'll get to that, your plan is to look at unprotected?

MR. WILLIAMS: Yes.

MEMBER ROBERTS: Okay, great. Thanks.

MR. WILLIAMS: Yes. Yeah.

MEMBER MARTIN: So, Eric, when you mention that, you know, your analysis shows you this large margin but with the, you know, mechanistic solution, of course we've not seen that. Right?

MR. WILLIAMS: Right.

MEMBER MARTIN: And I'm not sure where it is in submittal and review space. You know, we can only judge on what we've seen; right?

And traditionally DBCs have reason to fall back on. But the idea is that it comes first and, you know, and justification comes later, typically.

Now, in contrast, like the HTGR, you know, they had the advantage of all the, all the testing that was done at Idaho and kind of coincident with the writing of the Reg Guide. Of course, there was a, you know, fair amount of knowledge about how well TRISA performed. And that insight kind of fed the writing of that, of the Reg Guides. I'm tracking it a little bit.

But, you know, in other conversations I've had my understanding is that it influenced how, how that was written. With the SFRs you don't have enough read out there doing all this wonderful work for you. You don't have that kind of in the, in the, you know, public domain or, you know, working through it or see. It's all just to say justification hasn't come to us, and so it makes it very difficult for us to depart from what the Reg Guide says.

You know, and that was kind of the criticism that came up during the subcommittee. So, that's our, our perspective.

MR. WILLIAMS: We have some -- I mean, I'll point you to the Argonne National Lab did a trial mechanistic source term project in the ANL-ART series of documents. RT-3 is essentially a PIRT on mechanistic source term for Sodium Fast Reactor with metal fuel.

TerraPower participated with them in that, as well as other, other vendors. And we're heavily leveraging that work.

In fact, it talks a lot about the fundamental safety of metal fuel under, you know, a sub-cooled pool of sodium, and how it behaves, and the retention of fission products within the fuel matrix, the retention of fission products in the liquid sodium, and how all of that behaves. And adds a tremendous level of margin to safety.

And so, that's something in the public domain that I think is really good background to read. And then I think all of this will come together in the mechanistic source term topical report. So, that I think is coming up, so, can have a lot more discussion about that.

MEMBER MARTIN: I think you might get a different response if that came first.

MR. WILLIAMS: Yeah.

MEMBER MARTIN: Right? And it goes through the process, the sausage-making.

MR. WILLIAMS: Yeah.

MEMBER MARTIN: But in contrast, you brought this first. And so, it looks like the Reg Guide strictly applies without any further justification.

MR. WILLIAMS: Yeah. I can appreciate that.

MEMBER PETTI: I said it in subcommittee, you're not the first where because things are done sequentially, we're trying to see the whole elephant and all we see is a piece.

With other applicants, it wasn't until we got to the PSAR, where there's numbers in there and I went, oh. And the lightbulb goes off because you can finally see all the pieces come together.

And it's like, God, I ask all these stupid questions. I wish I knew this number when we started. It'd be more efficient in the overall process.

When we did this in NGMP, we asked a source to -- it was a white paper at the time -- and the fuel quality be done together, for exactly this chicken-and-egg problem, when you're dealing with new technology that's about a different education by -- it's just inherent in the new technology, I think.

MR. WILLIAMS: Yeah, these things used to be historically very separate conversations you could have. They were disjointed. And now, they're coming together with LMP, which is a benefit. But we have to have these integrated discussions a little bit better now.

All right, I'll talk a little bit about --

MEMBER ROBERTS: Probably leads to what you're about to say. One big-picture question comes up with the Reg Guide at 1.232 and the fifteen-containment criteria, that resulted from decades of progression -- if you look back at history, I'm sure you're versed in all of this -- but going back in time to S-PRISM, PRISM, IFR, go back, and all the different developments in studying past reactors, there was a 1993 proposal to ease up on some of the requirements, and you could argue that the results of that led to a risk-informed, performance-based, set of containment criteria for an SFR.

And that's all got codified in Reg Guide 1.232. So, it's kind of hard to see the case for diversion from that given all the history, and the fact that your client looks a lot like PRISM.

So, the accumulated judgment of all those different generations of designers, led to those criteria. So, maybe you can lead into your discussion here with that as the starting point, is that when you make what might be radical changes -- and that's my question would be, are they really radical changes, because I'm not quite sure they are -- but what appear to be radical changes, then the justification overall for what's different now, than what was in the previous designers' minds, all those generations of SFR developments, is kind of a question.

The second question is are you really being different, because there are aspects of your design that look a lot like the containment approach for PRISM.

And so, it seems like you're actually including the structures that PRISM had to meet their containment objectives. And so, if that's the case, then what's the implication of changing the criteria?

MR. WILLIAMS: Okay. Yeah, that's a great segue. I was going to talk through some of the design features that are different. There's essentially five areas that I wanted to bring up and just talk through a little bit.

And some of these are radical departures from historical sodium-fast reactor designs. I'll try and highlight those in particularly.

But first of all, this is not a departure. But just the use of sodium coolant, of course, is just a huge benefit in terms of having a low-pressure system that doesn't have the forcing function for radionuclide releases through the functional containment, the maintenance of highly sub-cooled sodium within the reactor vessel to retain fission products -- also very important, so that's a part of all sodium-fast reactors.

The fact that it's a full metal-fueled core is also a departure. I think we are the first fully metal-fueled cork. So, that takes away a couple of things that were being looked at from the hypothetical standpoint, that involved core disruption-type accidents.

So, oxide fuel behaves very differently in severe accident space than metal fuel. And of course, having an integrated reactor vessel with a large pool of sodium also makes the bulk boiling of the sodium an incredible event.

So, that, the maintenance of a highly coolable geometry for metal fuel, that's one of the departures.

The metal fuel also retains the fission products in fuel matrix -- certain categories of the fission produces, I should say.

And then having a pool reactor is another big one. Because it drastically reduces the amount of sodium piping that you have for the potential of sodium fire.

So, keeping all of the primary system piping inside the integrated reactor vessel, you've removed the fundamental hazard, which is the best thing you can do.

Then, what you're left with is the sodium processing system, which is small-bore piping that is contained within the functional containment boundary. And I'll explain in a little bit how we address sodium fires in a different way there too.

Because the bulk boiling is not a credible event, what we are looking at is the potential of gas bubbles in the fuel channel from fission product release from a failed fuel pin. So, we have to look at that.

We're looking at potential entrainment of cover gas as a way to get bubbles into the core. I think that's highly unlikely to happen in such a large pool reactor, but we are addressing that as well.

And when we look at that in our analysis, we really don't see anything that would propagate fuel pin failures within the assembly.

So, when you have a fission gas bubble going up through the channel, the temperature of the neighboring fuel pins barely increases. So, that is looking to be a really good analysis.

So, we're not seeing any way for voids to cause a large energetic release. So, that's all coming from the fact that there's a pool reactor here.

And then one of the really big departures is that there's no longer a sodium water steam generator. So, that was one of the huge sources of energetic release that pressure-retaining containments had to address.

And so, by having molten salt energy storage and having an intermediate sodium system, we've eliminated that hazard from the design. So, that's a key one.

We're also excluding any kind of water suppression systems from inside the functional containment, and being very careful not to introduce water even in that space. So, addressing it there.

And then in terms of sodium fires, a lot of other designs have used what appears to be similar design features -- guard pipes, and things like that, around sodium pipes -- but we have addressed that entirely for the functional containment space, by having a secondary barrier around all sodium piping within the functional containment barrier, even the intermediate sodium piping that's in that space above the reactor. So, we're addressing that.

And then we have the guard vessel that surrounds the reactor vessel. So, even an unlikely leak from the reactor vessel would be contained in an iterative space there.

So, by addressing these features in the design, we've essentially eliminated those large energy releases from the functional containment. And as it talks about in the SECY paper, we really have all of those conditions that would make a functional containing approach fit.

We have a new coolant, we have a new operating state, a close-to-atmospheric pressure, and we've removed a lot of the major accidents, like seeing generator and sodium water interactions.

So, that kind of covers the design features. And then talking a little bit more about how the process looks when you're designing the functional containment and setting the performance criteria to take all these things into account, like I said, it's really integrated together with LMP and with SARDLs.

Because, essentially, the LMP is used to establish the LBE categories that you're looking at. And then the functional containment performance criteria is established for each of those LBE categories.

So, that includes the SARDLs. The SARDLs are included for the normal and AOOs, 10 CFR 5034 for the d/b/a's, etc.

And then what you do is you establish clear barrier performance criteria for all the SSEs in the functional containment. So, again, that goes back to the design process and integrating the safety requirements with the design from the beginning, and being able to review those over and over again as you go through the LMP process.

And then we do demonstrate that those performance criteria met with a major accident. And if you want to reference back to the July 2023 meeting we had with the NRC staff, they addressed SARDLs, functional containment, and the major accident, and kind of laid out the different major accidents we would be using, and some of the main assumptions going into those.

And then finally, the mechanistic source term then pulls from all of these areas, to actually go through and demonstrate the safety margin.

And then what you see, like I said before, the specific PRA results on a frequency consequence curve.

So, that's how functional containment works with all the other elements of the LMP.

MEMBER ROBERTS: I have two questions of what you just now laid out. Of all the features, it seemed like all of them are also characteristic of PRISM, except for the steam generator they moved outside of the containment.

So, is there enough of a difference from PRISM that the thought process that went into the PRISM containment approach is no longer needed?

MR. WILLIAMS: In my opinion, yes. I think we've also gone further with the sodium fire protection within the functional containment space, than was done in PRISM. I'd have to go back and check that.

But the steam generator removal is a pretty huge one, I think. And I think applying it with LMP is also different, because the LMP gives you the hooks, if you will, to demonstrate functional containment, along with all the other safety features, throughout the design process.

MEMBER ROBERTS: If you look at the approach PRISM used to containment, it was a guard vessel, and I guess they call it the containment dome, which is the equivalent of your -- area access enclosures, as I read it.

And looking at the PSAR that staff just accepted a couple of weeks ago, one of the criteria that you list is maintain at least one barrier between the clotting, piping, or vessel, containment-ready nuclide source to withstand all the design basis access conditions, and whose leakage is specified by design requirements for testing, which sounds a lot like the PDCs are going to take them out.

To have a containment structure, to have leak testing, design requirements that ensure that the leakage rates are met, that seems a lot like the containment requirements that are in the SFR design criteria.

Again, that's not to say -- it's just a very radical change. You talked about having some sort of containment around the intermediate sodium piping. Is that the head area access, or is that something else?

MR. WILLIAMS: Yeah, essentially, for the core, the primary safety-related boundary essentially is the reactor coolant boundary. And then the secondary barrier is essentially the guard vessel and the head access area combined.

So, those are the boundaries that you think of for functional containment. And so, essentially, what we would do is meet that criteria through the GDZs on the primary coolant boundary, that are essentially very similar to that.

MEMBER ROBERTS: All right. So, as it seems almost like rearranging the deck chairs, that either you have the same containment capability, it seems like that S-PRISM, PRISM, the plants that were the foundation of the SFR design criteria, you have that, you're going to have to have design criteria to show you meet them, which it seems like those are the design criteria that are specified for SFR design criteria in the appendix of the Reg Guide.

So, again, I'm just trying to understand, are you really doing something radical, or is this really just making the terminology map up with LMP?

MR. WILLIAMS: I don't think the head access area is a pressure-retaining containment. So, the fact that we don't have the sodium water reaction in that space to require that is probably the main difference there.

MEMBER ROBERTS: Right. And if I remember at the SFR design criteria, it allows you to figure out what relatively low pressure you would need for that pressure-retaining containment.

So, that's already one of the performance-based allowances in the Reg Guide. Again, maybe it requires some more thought, but it seems like what you're doing isn't necessarily, from a design perspective, much of a change from PRISM and what the SFR design criteria are trying to push.

In which case, maybe you'll end up putting them back in. I don't know, I'm just trying to understand. But seems like you would need testing requirements for leakage if you have a requirement, self-imposed, that you have leakage specified by the time requirements for testing.

MR. WILLIAMS: Yeah, I assume we would be meeting all of those requirements and demonstrating that anyway, even without the criteria in there.

MEMBER ROBERTS: Okay. Yeah, I think that makes sense. Thanks.

MEMBER PETTI: I just think LMP provides a structure.

MR. WILLIAMS: Yeah. Mm-hmm.

MEMBER PETTI: That if you develop these design criteria, let's say from the bottom up and years of experience, LMP gives you kind of a top-down way to look at it and make sure you meet in the middle.

And we kind of assure you there's designers early in the process, that the requirements have the right, the requirements at a broad system level, that in principle, going bottoms-up you could miss something. Right? And then go, oh yeah, down here we got to go backtrack.

LMP, if it's done iteratively, like it says, prevents or minimizes that sort of backtracking.

MR. WILLIAMS: Yeah. Yeah, that's right.

MEMBER PETTI: Yeah.

MR. WILLIAMS: That's right. I think it makes the conversation clearer, and I think there's a lot of value in that.

MEMBER ROBERTS: Did we already cover what you planned to present?

MR. WILLIAMS: Yes, yes, we did, I think.

MEMBER ROBERTS: Okay, great.

MR. WILLIAMS: Yeah, we did.

MEMBER ROBERTS: Very helpful.

CHAIR KIRCHNER: Could you just flesh out for us an example -- let's pick on something that you already identified as one of your design features and one of your barriers. What would the performance criteria look like for the guard vessels?

MR. WILLIAMS: Yes. So, there's a couple of key criteria on the guard vessels. So, essentially, a postulated leak from the reactor vessel has to be contained within the guard vessel, and the gap within the guard vessel is a size such that it remains above the heat exchangers and the reactor vessel and the pumps so you can continue to provide aquicore cooling.

Yeah, it has a function there. It also carriers a radionuclide retention function as a secondary barrier for the functional containment, the primary barrier being the reactor vessel.

So, if you assume the fuel pins have failed -- and we assume all the fuel pins have failed -- and demonstrating this, we assume the failed barrier there, if you also have a failure in the reactor vessel barrier, then the guard vessel is there to prevent further leakage.

CHAIR KIRCHNER: So, effectively, when you implement these requirements, you're going to have something that -- don't like to use the LWR terminology -- and essentially leaked barrier about, if not even a more demanding requirement, regardless of the fact that you have sodium --

MR. WILLIAMS: In some cases it is more demanding, because we're trying to prevent the sodium from contacting air as well. Yeah.

CHAIR KIRCHNER: Members?

MEMBER ROBERTS: Okay, it sounds like we've no more questions. So, thank you very much for your presentation, TerraPower, and I guess we'll switch to the NRC staff now.

(Off-mic comments.)

MEMBER ROBERTS: Then let's take a break until ten o'clock, giving the staff a chance to set up.

CHAIR KIRCHNER: So, for those online listening in, we're going to take a break until ten o'clock, Eastern Time.

(Whereupon, the above-entitled matter went off the record at 9:50 a.m. and resumed at 10:01 a.m.)

CHAIR KIRCHNER: Okay, we're back in session. And I just want to go ahead and turn it back to Tom. Go ahead, introduce the NRC.

MEMBER ROBERTS: Thank you all. And I'm just going to go ahead and pass it over to the NRC staff. Mallecia, are you going to start? Or Stephanie?

PARTICIPANT: Actually, we're going to have Reed start it.

MEMBER ROBERTS: Reed start it. All right.

MR. ANZALONE: I'm just going to take it from the beginning.

MEMBER ROBERTS: All right, go ahead. Go ahead, Reed. Thanks.

MR. ANZALONE: Thank you, Member Roberts. So, I will jump straight into it. We had most of the members for the subcommittee meeting. So, I think the goal is just to try to cover the key points from that subcommittee meeting. And I think TerraPower did a good job of laying out a lot of the technical aspects related to PDC. So, we're going to basically just focus on our approach for the review.

So, I'll briefly cover the purpose of the topical report and our strategy for the review, talk a little bit about the regulatory requirements, give a real brief overview of the PDCs, and then jump into key topics from the subcommittee meeting, which are functional containment SARDLs and the limitations and conditions. And the slides aren't advancing.

Okay. So, the purpose of the topical, like TerraPower talked about, was to describe the process for developing PDCs, and then actually give us those PDCs. And that's partially to address compliance with 10 CFR 5034.

They also wanted to describe their rationale for meeting the intent of PDC 26. I'm actually not going to talk about that today, just to be clear, to focus on the topics from the subcommittee meeting.

And then our strategy for the review was to review the PDC's conformance with the Reg Guide group and evaluate the deviations from the Reg Guide, considering the key design features.

And really, our scope, we wanted to -- and this is something that we struggle with a little bit as the staff for PDCs, because it's very easy to get into the technical details of how you're going to comply with the PDCs -- we wanted to focus on whether the PDCs themselves were acceptable.

And the design is an appropriate and necessary context for that, but we didn't want to get too into the weeds on how they were going to meet them.

Part of our review then also was identifying the interaction between the Reg Guide 1.232 approach and the LMP, which TerraPower talked about a little bit today. And then the PDC 26 rationale was a specific subject that we tackled, that, again, I'm not going to talk about really today. Apparently, I don't know how to move the slides forward.

All right. Okay. So, the regulations, I already mentioned that 5034 requires the CP applicant to include the PDCs. TerraPower had the topical report, which I believe is incorporated by reference in the PSAR, but then they also put the PDCs into the PSAR as well. But this was submitted well in advance of the construction permit application.

And then Part 50, Appendix A, which has the general design criteria, provides requirements on the scope and content of PDCs, for all reactors, including non-light water reactors. So, that first bullet there talks about what the PDCs need to be able to do.

And then the second bullet is really sort of more guidance saying that the GDCs that are in Appendix A provide guidance for how the PDCs should look.

So, TerraPower developed their PDCs, like they mentioned, based on Reg Guide 1.232. Most of their PDCs were directly based on the SFRDC, which were in Appendix B of that Reg Guide.

Some PDCs were based on the modular high-temperature gas reactor design criteria, which are in Appendix C, and those were generally used to implement functional containment, or reflect the use of SARDLs.

Most of the PDCs were modified in one way or another from the base design criteria in the Reg Guide, and we kind of circled around it a little bit in the conversation earlier with TerraPower.

But there are no design criteria for those numbers down there, due to the use of functional containment. Talk about that a little bit in the next couple of slides.

So, these were the general changes to PDCs that I laid it out in subcommittee meeting. We're going to focus on those two today, just for the sake of keeping things a little tighter.

So, starting with functional containment, so, obviously, discussions on containment -- I think members have mentioned this earlier in the meeting -- the discussions about containment and functional containment, and what's appropriate for NSFR containment, have been going on for a long time.

Part of that discussion is Reg Guide 1.232, which has containment criteria in it for SFRs and for MHTGRs.

But one thing I will say is that Reg Guide 1.232 came out, and then SECY 1896 came out, which actually sort of codified the functional containment approach.

And so, I'll talk about that more on the next slide, our take on that SECY paper and the associated SRM.

But sort of even at a high level, Reg Guide 1.232 talks about functional containment. Yes, it's in the MHTGR DC, and I think a lot of the impetus for developing that concept came from the HTGR world and TRISO fuel.

But the approach is technology-inclusive. And the Reg Guide says it's applicable to advanced non-light water reactors without a pressure-retaining containment structure.

So, our thinking -- and I'll go to the next slide to talk a little bit about the SECY paper -- is that from the get go as part of the current conversation on functional containment, the idea is that it is technology-inclusive, risk-informed, and performance-based approach to containment design criteria.

The SECY paper, which was approved by the Commission, gives a methodology for determining functional containment performance. That developed into LMP, and it was developed in parallel with the Reg Guide, which noted that some of the stuff still needed to be approved by the Commission.

MEMBER ROBERTS: Maybe you could comment on -- I'm going to make an assertion and you can tell me where I'm wrong.

It seems like the Appendix B -- SFR design criteria for containment -- are technology-inclusive for an SFR, risk-informed, performance-based, because if you look at the history, it seems like the NRC took a turn at that probably 30 years ago, and said, we need to go revise the GDC that are derived from light water reactors, because they don't really apply to this technology, and what's left there does have some aspects of at least performance-based and, I'd expect, risk-informed.

So, is it fair to say that the existing SFR design criteria for containment are risk-informed, performance-based?

MR. ANZALONE: I would say that that's true to a degree. And I think I -- and maybe it's the next slide -- I'm going to talk a little bit more about the specific SFR design criteria, which do have kind of notes in them about, this would apply under these certain situations.

But at the same time, if you go back and look at the SECY paper, I had the benefit of going through the transcripts from the ACRS meeting and your letter on this, and I think sort of conceptually, the thing that functional containment as an approach does, is it's capable of encapsulating all of the possible different approaches.

So, if you look at SECY, I think it's 93092, which might have been what you were talking about 30 years ago.

There's a bunch of different containment designs that are referenced in that. There's the MHTGR, which is sort of a pure functional containment along these lines.

There's the PRISM reactor, and I think it was the OR S design. That all have kind of varying degrees of containment, leak tightness, and different containment designs.

And if you look back at the SECY paper and sort of -- a bunch of the discussion around that, the idea was that functional containment performance could be, you could define a generic functional containment criteria that could encapsulate all of that.

So, I think the approach that was in SEC 1896, is intended to kind of wrap around all of them. And I think the letter that the ACRS wrote at the time actually kind of explicitly says, hey, maybe the staff should go back and revise Reg Guide 1.232 to say, hey, this concept could apply across the board.

So, that was part of our consideration here in thinking through does functional containment make sense for TerraPower?

CHAIR KIRCHNER: Reed, let me help you here. I have the letter.

(Laughter.)

CHAIR KIRCHNER: And it goes on to say that the containment criteria in Appendices A, B, and C of the draft Reg Guide are logically inconsistent. So yes, there was this thought that they should be technology-inclusive.

So it wasn't clear at the time whether the staff would go back and revisit the Reg Guide. I think there was a pointer there that the sets of criteria weren't, as we stated, logically consistent.

MEMBER ROBERTS: Yeah, and we're 30 years later and this is Natrium, not PRISM, and there's a lot of development of risk-informed thought processes. And that can certainly result in another term.

But it seemed to me, I'd want to see, get your reaction, that the existing Reg Guide Appendix B is risk-informed, performance-based for that specific technology. And that doesn't mean that's set in stone, because there are changes to, you know, from PRISM to Natrium, and there are changes in thought processes, or risk-informed space.

And it seems like okay, an unfair statement to say this is the addition of a functional containment thought process, because you could argue, you know, that term wasn't used. This was essentially a functional containment for and SFR developed 30 years ago, just not using that term. Is that fair?

MR. ANZALONE: Yeah, and I can -- I just moved on to the next slide, because I think this sort of talks about what you're getting at. There are certain of the SFRDC that talk about, you know, how they would be applicable if certain approaches to containment were taken or might not be -- like maybe you would have the SFRDC, but it wouldn't actually be applicable to any structures at the plant, which is kind of an odd thing.

So that's the, basically I think it's, yeah, 39 -- 38, 39, 40, and 50-57 are all sort of in that space where, you know, you could maybe make the argument, okay, we don't need this structure, even though we have these criteria. But it, to me, it's not, that's not like a clean approach. That's messy.

And I understand that, you know, the SFR or the functional containment criteria is itself a little bit messy because we kind of get everything at once with the demonstration of functional containment performance. But the criterion itself is more straightforward, and you don't have to make these arguments about how we have these criteria but they're not actually applicable.

So like for example, if they didn't need heat removal in the containment, then they wouldn't do anything with 38, 39, or 40. But then why do you have them at all? And to me, it makes more sense to apply like a sort of more straightforward performance-based criterion that encapsulates everything.

So our take is that it's, you know, appropriate to apply the functional containment criterion for TerraPower. And I'll, I guess I'll go onto the next slide.

Because you know, the SECY paper talks about it being acceptable for non-light water reactors. We do think, and TerraPower I think laid these out very well, there are attributes of the reactor design that are necessary to be able to, you know, effectively actually use a functional containment approach.

But that functional containment performance still needs to be demonstrated, and that's part of our review in the construction permit application. And you know, based on what I read in the transcripts and the discussion surrounding this issue back when ACRS reviewed it back in 2018, there were a few sort of thoughts about, you know, defense-in-depth and how you would go about actually analyzing those.

So I just wanted to throw these points in here that LMP is really like a key part of this. And it's part of why we have that limitation condition that says thou shall use LMP is you're going to apply this approach. It implies that you're going to have a PRA and a mechanistic source term. It gives you criteria that you need to meet.

You have to explicitly consider uncertainties. And you have to do this risk-informed, performance-based, defense-in-depth adequacy evaluation that's in NEI 18-04. So you know, functional containment doesn't mean no containment, it means you evaluate all of the barriers that are in the way of the release of radionuclides.

So if we were going through TerraPower's evaluation in our review and we came across something that we felt like releases weren't being appropriately addressed, that's something that we would bring up during our review.

Okay, any questions? Because I'll be moving on to SARDLs, which that's a pretty brief discussion.

So SARDLs were initially identified for TRISO fuel for the MHTGR. They're for normal operations in AOOs, and they need to be established so the Part 20 limits aren't exceeded. But the SECY paper on functional containment performance criteria does pretty much say SARDLs are intertwined with functional containment performance criteria.

And so, you know, I think the concept that we understood from talking with TerraPower about this is that the impetus for SARDLs is that use of functional containment, which is also all intertwined with LMP.

And so that first bullet here on this slide is that we -- our view is that SARDLs are appropriate (audio interference) and consistent with a performance-based evaluation.

We already talked about fuel design limits that can be used to help evaluate those SARDLs. And I'm really glad that Eric touched on it during his presentation. I think one of the key things is that SARDLs are a useful tool for looking at ex-vessel events and sources of radionuclides other than just the fuel inside the reactor.

And he mentioned that ANL art series of reports looking at mechanistic source terms. One of the things that ANL has found, and I think TerraPower's assessment also agrees with this, the things that drive the plant risk are not the in-vessel events. They're all of the issues in these like auxiliary systems and fuel-handling accidents.

So having SARDLs to evaluate those events actually helps a lot. Part of SARDLs is that you would need to include a means of monitoring activity in these systems, so that's something that we'll evaluate as we look at their plant design.

And the SARDLs need to still be proposed and evaluated, and we did discuss them. Eric mentioned the July public meeting, where we had some example SARDLs that we talked about with them.

Moving on to the scope and applicability of PDCs, and really this I just wanted to say, you know, we talked about possible changes to the limitation 2 and RSE. As of right now, we haven't identified any changes, and so we didn't pass along to the ACRS. So that's why I wanted to bring that up again in this meeting. So that limitation 2 really is focused on the use of LMP.

And these are the same conclusions from the subcommittee meeting, so I won't reiterate them in the interest of trying to get us closer to the schedule in the agenda, so. Anyway, happy to take any questions that you may have. If not, then we can --

CHAIR KIRCHNER: Reed, when you went through this, okay, so you accept the premise. Did you systematically look at the implications of expunging, or maybe a better way to say it is to divert from the ensuing GDCs that are containment-related? See where I'm going with this?

If you say 16 is now functional containment rather than containment, then do you still systematically look at all those other GDCs that support containment? Because what they by and large do is protect that fission product barrier in once sense or other. So was that --

MR. ANZALONE: So yeah, there a couple --

CHAIR KIRCHNER: What you're thinking?

MR. ANZALONE: Yes, and there are a couple of PDCs that TerraPower added back that talk about the performance of the reactor building envelope and stuff like that are -- that are necessary when you use a functional containment approach.

CHAIR KIRCHNER: You feel that you've got a complete set and that would address those other functions that, how should I say this, that the containment building structure provided, went beyond just fission product release. Either they were protecting the building de facto for a LWR becomes the major protection against external events, or many external events, etc.

So the containment function goes beyond just fission product barrier purposes.

MR. ANZALONE: Yeah.

CHAIR KIRCHNER: Protecting against

internal, external events, flooding, fires. So when the staff did its review, you felt there was a complete set of those other functional attributes that go into -- unfortunately, containment for an LWR is a multipurpose --

MR. ANZALONE: Right.

CHAIR KIRCHNER: Function. So you're satisfied that they address those -- their thing.

MR. ANZALONE: Yeah.

CHAIR KIRCHNER: So when we come back and look at an actual detailed design and look at, let me pick one of the things that's always problematical with containment is double isolation valves, inside, outside, and so on.

You still feel that the functional equivalent of those containment-like criteria would still be applied when you reviewed individual fission product barriers, i.e., a guard vessel?

MR. ANZALONE: Yes.

CHAIR KIRCHNER: Think someone was just unmuted there. Okay.

MR. ANZALONE: But yes.

CHAIR KIRCHNER: All right, thank you.

MEMBER ROBERTS: Yeah, I guess I thought of it a little bit differently. Because the functional containment is not entirely defined yet, if it's defined as the PSAR indicates with containment structures and leak test requirements, and the like, you would add back in criteria as the design of the containment would like more and more like a classic SFR containment or light water reactor containment.

That, that was my interpretation as you got the big picture, you know, this is a functional containment, we're going to figure out what it means. But then when it looks like a more conventional containment, you have to look at putting back in these kind of design criteria.

Whether they're called PDCs or what you call them I don't know, but I would think you'd still want to make sure the requirements for leak testing or the requirements for double valve isolation, whatever they happen to be, are met once the containment structure looks like a structure that these were applied to.

MR. ANZALONE: I'm not sure I followed.

MEMBER ROBERTS: Yeah, that's still puzzling me a little bit. The proposed containment for the PSAR is to have a structure, right. So there is a structure around each boundary that contains reactor material, which ends up looking a lot like, you know, a PRISM containment.

And so it looks like a PRISM containment, then the requirements that were based on PRISM containment will seem to be met in some form or have to be met.

And so whether you call those PDCs or call them design requirements or tech specs or whatever they happen to be, once you -- once they go back to the design looks like PRISM, then the requirements are imposed in PRISM structure would seem to be evaluated, need to be evaluated for applicability and things like leak test capability would seem to need to be a requirement. Just like PDC-52.

MEMBER ROBERTS: So some of those detailed design requirements would, you know, depending on the specific design of the system would I would expect sort of flow down from the high level performance requirement in the PDCs.

CHAIR KIRCHNER: So where I was coming from is if you were to go back and look at that -- and I'm sure you have -- the staff's work on functional containment, they point to additional sets of functional containment performance standards, like protecting other risk-significant SSCs.

MR. ANZALONE: Yeah.

CHAIR KIRCHNER: Support of them, occupational radiation exposure, removing heat, physical protection, like security for external events, etc. So that's where I was going.

MR. ANZALONE: Yeah, that's what I understood from you. From Member Roberts, I thought you were talking about sort of more detailed criteria for specific system designs. Is that correct?

MEMBER ROBERTS: Yeah, I'm thinking if these 15 criteria that are in Appendix B or based on the characteristics of a PRISM containment structure, and if the nature of a containment structure looks like a PRISM containment structure, then the same 15 requirements would seem to need to apply in some form.

Whether you call them derived requirements or principal design requirements or whatever, if the containment structure requires them to meet its function, then they would need to be tracked I would think in some form.

MR. ANZALONE: Well, I guess I would say that with that high-level performance-based requirement, you know, TerraPower would look at their design and they would look at the releases and the doses, and they would figure out which of those kinds of criteria would need to be looked at.

So if you look at like double isolation across containment, I don't -- I don't think that something like that is part of TerraPower's design. I don't know for certain, and I'm sure it would depend on the specific system. So that's maybe an example of where like using a performance-based criterion would buy you something in design space.

MEMBER ROBERTS: And if just folks want an example, Criterion 52 says a reactor containment structure and other equipment that may be subjected to containment test conditions shall be designed so the periodic integrated leak rate testing can be conducted to demonstrate resistance and containment design pressure.

So if their design is going to have a low leakage, you know, structure for the -- the head access area, and their intent is to verify that by test, then why wouldn't 52 apply?

MR. ANZALONE: Because it's encompassed by this functional containment performance criterion.

MEMBER ROBERTS: So it ends up being a derived --

MR. ANZALONE: Yeah.

MEMBER ROBERTS: Not a principal design requirement.

MR. ANZALONE: I would agree with that.

MEMBER ROBERTS: Okay. But in some form, it would seem like your review of that system, once it's concluded that it looks like PRISM --

MR. ANZALONE: We would --

MEMBER ROBERTS: You would --

MR. ANZALONE: We would want to make sure that it met its performance requirements. That the performance requirements made sense and that it met them. But that is all down the road as part of our construction permit application review.

VICE CHAIR HALNON: But so -- this is Greg. Those performance requirements, I mean, to your point, if it's not required for part of the functional containment definition of whatever that SSC, if you would, then you wouldn't have to do the leak testing. This really doesn't apply.

MR. ANZALONE: Right.

VICE CHAIR HALNON: It's not part of the basis for that containment. I mean, so it's not really "containment structure." That's the way I'm reading that. I don't have a conflict here.

I see it -- see what -- the functional containment, you could look at that as an SSC, even though it's distributed amongst things. This is not part of that SSC. That's the way I'm looking at it. So I don't see a conflict in my mind.

MEMBER ROBERTS: It depends what is in the system to meet the functional containment requirements.

VICE CHAIR HALNON: Right. So if that vessel, guard vessel, you say is not supposed to be, you know, a gaseous leak-tight, if you would. It's sodium leak-tight. It doesn't make -- in my mind, it's just not part of that functional containment requirement. So there's no -- 52 wouldn't apply. Even though you say it's derived, it's sort of derived. But it's not part of the SSC for functional containment.

I'm not looking at specifically design, I'm looking at conflict. I understand how 52 would not be part of this because it's not part of the SSC of functional containment, in a classic sense.

Anyway, I just wanted to make sure that I understood why you said it was derived. No leak test is required because it's not part of the SSC.

MEMBER ROBERTS: Yeah, it depends on the functional containment model on what they've got in there. Well, what they say in PSAR is to have these structures surrounding the vessels and pipes and the like that contain radioactive material. And they plan to leak-test everything that's in that.

So the question would be whether that derived requirement ends up being something that the NRC staff would validate because that's part of the approach taken to containment that looks like PRISM, or something that's just part of developing the functional containment model and whether or not it's derived from that.

I'm not sure that distinction is clear, but if the structural containment looks like PRISM and it's credited, you know, similar to the way it was operated in PRISM, then the requirements there were applied to PRISM would seem to apply also.

VICE CHAIR HALNON: Unless they're subsumed into something bigger.

MEMBER ROBERTS: Yep.

Any other questions for Reed on this subject?

CHAIR KIRCHNER: Reed, at some risk I'm going to bring up PDC-26. I didn't want you to get off that easily. Can you just for the record, since this is full committee, give us your evaluation of the proposal for PDC-26?

MR. ANZALONE: Yeah, sure. So, and we did cover this at the -- I don't think I'm going to say anything different than I said at the subcommittee.

CHAIR KIRCHNER: I don't expect that.

MR. ANZALONE: For the record, you know, TerraPower proposed that they would essentially adopt the SFRDC-26 with some conforming changes about safety significance that are consistent with LMP that are applied to all the different PDCs. So it's essentially, I would say it's essentially unchanged from SFRDC-26 in like a meaningful way.

And so then they proposed that they would meet that by essentially showing they have two different -- two different control rod designs that mitigate common cause failures between the different control rod designs.

And so they were intending to show that there was sufficient diversity and independence between the different control rod design. And there's a different means of control rod insertion. So they wanted to show that in a sort of risk-informed manner, that that would be independent and diverse enough to meet Criterion 26.

CHAIR KIRCHNER: But the staff position, as I understand it now, is you basically accept that, but it's TBD --

MR. ANZALONE: Exactly.

CHAIR KIRCHNER: D being demonstrated by design that they're not going to be subject to a common cause failure.

MR. ANZALONE: Correct.

CHAIR KIRCHNER: Seismic misalignment such that you can't insert either set of control rods. So this, I'm just flagging this because it's a major departure from what in the past had been the definition of diverse, looking at two diverse or similar systems for that function.

MR. ANZALONE: Yep. One thing I will say fortunately that SFRs buy you, and I know this was mentioned during the subcommittee meeting, you know, you can fail a lot of control rods and still get enough negative reactivity insertion to shut down the reactor. So I think that would help with the overall demonstration, that there's enough diversity there.

CHAIR KIRCHNER: Perhaps the exponents of the opposite take would say with an SFR, you can get a significant reactivity insertion event.

MR. ANZALONE: Absolutely, and that's something we're going to be really focused on in our review.

CHAIR KIRCHNER: Yeah. Because the one

thing, if we're here I'll just state this, is that we're now at a size for a fast reactor design like this, that you're at the edge of where you can rely on the leakage as a negative feedback mechanism. So I expect that when we review the PSAR, that this will be looked at very carefully when we're considering the PDC-26 as well.

MR. ANZALONE: Absolutely, absolutely, I totally agree.

MEMBER ROBERTS: Any other questions from the members or consultants on the PDC? Now we can move on to the fuel qualification report.

MS. DE MESSIERES: Actually, this is Candace de Messieres from the NRC.

So I just wanted to make one clarifying point for the record as it relates to L&C No. 2, that at the highest level, that L&C has to do with the synergy between the frameworks between PDC and LMP. And that the staff continues to work to ensure clarification at a generic level on that issue.

So I just wanted to make that note for the record. Thank you.

MEMBER ROBERTS: Thank you.

MR. ANZALONE: Okay, so moving on to fuel and control assembly qualification. So we started with fuel and we'll end with -- we'll let fuel take us out.

This is a, I would say, more just relatively straightforward truncation of my presentation at the subcommittee, so I can go through this as quickly or as slowly as we want. So I'll talk a little bit about, again, the topical report purpose and our strategy in the review. I'll talk a little bit about the regulatory requirements in the guidance.

And part of what we used a lot in our review was this NUREG-CR 7305 for giving us technical information that we could use to help evaluate TerraPower's fuel. Then I'll go through a brief overview of our safety evaluation and the overall conclusions.

So the purpose of the topical report was to provide a plan to qualify Natrium Type 1 fuel, which as TerraPower talked about, is a U-10Zirc metallic fuel in HT9 cladding. And they're control assemblies. And it requested NRC review and approval of a bunch of different items that essentially are the fuel qualification plan.

And it provides some fuel qualification results and talks about their ongoing plan of fuel qualification activities.

So our strategy in the review was to review the scope and adequacy of the plan in the context of NUREG-2246, which was released after TerraPower had started developing this topical report. So they included a crosswalk that sort of referenced their criteria that they came up with against NUREG-2246.

And then we also reviewed it, as I mentioned, against NUREG-CR 7305, which you know, I should say is not, it’s not -- it doesn’t have like the status of guidance, right. It’s not a reg guide. But it is additional technical information that we had contractors from several different national labs put together to help us look at metallic fuel.

So the regulatory requirements. And NUREG-2246 I think does a pretty good job of laying out the landscape of how fuel qualification works in terms of regulatory requirements. It provides a lot of the technical basis for how you would show that you meet the regulatory requirements.

But there aren’t necessarily a ton of regulatory requirements that directly apply to fuel qualification as a process. But 50.43E requires your safety features to be supported by analysis testing operating experience of a combination thereof. And it requires there to be sufficient data to exist to assess your analytical tools.

And then 5034 requires applicants to evaluate a postulated of fission probabilities from the core in the containment. Part of that is the fuel's performance. And as TerraPower mentioned, that is something that they are doing with their major accident as part of their construction permit application.

And it requires the principal design criteria to be submitted. Some of the PDCs have to fuel, so.

So then the guidance, there's NUREG-2246, which provides general guidance on fuel qualification for non-light water reactors in the form of this fuel qualification assessment framework. And I'll be kind of stepping through that a little bit today.

And that kind of, that draws on a lot of the experience from the staff evaluating both light water and non-light water reactor fuels. And then also we have this NUREG-CR 7305, which was developed by staff from, I think it was INL, Los Alamos, and ANL, giving us some insights into metallic fuel systems.

And it did that in the NUREG-2246

framework and identified an operating and low-key behaviors and phenomena and provided a review of the data that was available and discussed a little bit the current state of fuel-performed follow-up.

And some of the key conclusions from the NUREG-CR, I won't go through the whole thing in detail because I did that during the subcommittee, and it took a solid 20 minutes. But for fuel with geometry and operating conditions consistent with the previous operating experience, so that's really EBR-II and the FFTF, MFF fuel, the metallic fuel that was operated at FFTF.

The life-limiting and safety-related fuel behaviors and well known and predictable, up to around 10 percent burnup. And that's not really a hard limit, that's a, you know, we think it's well-characterized up to this limit. Somewhere beyond that point, the behaviors are less predictable. And so if you wanted to go much beyond that, you would need to do a more thorough job of characterizing it than has been done previously.

Fuel constituent redistribution is one of the behaviors that is present in the data that does affect fuel properties and other things. That's captured in the existing empirical models that are based on this fuel operating experience. The life-limiting phenomenon is fuel cladding chemical interaction, which TerraPower mentioned as being their main issue that they're dealing with.

And fuel cladding mechanical interaction is not really a concern. But again, that's really specific to, you know, similar geometry to what the previous operating experience was, and but the lower end of the burnups that were operated.

Transient data would help to establish safety margins. TerraPower talked about doing additional transient testing. And that if you wanted to use a highly mechanistic model, you would need to do more work to qualify that. So for example, you know, what effect does fuel constituent redistribution have. That's something that you would need to study a little bit more closely.

So the Natrium fuel assembly design, it's very similar to the EBR tool and that MFF fuel from FFTF. It's a U-10Zirc peak enrichment less than 20%, so it's HALEU. Seventy-five percent smear density. These are all essentially the same characteristics that are discussed in the NUREG-CR.

TerraPower showed the assembly overview. You saw the hexagonal fuel assembly. And then they're applying this limited free bow core restraint system, which was sort of tried at FFTF. Or I guess was, you could say was tried at FFTF. So there is some information that they can validate against there.

Now, I'll just start walking through the fuel qualification big framework from NUREG-2246. So just it has this top level goal that fuel is qualified for use, and that's supported by all of these different subgoals. So Goal 1.1 and 1.2, or really all of Goal 1 is talking about the fuel manufacturing and whether that's in an appropriately controlled and understood process.

Our take on all of this was that the TR either includes or refers to design documents that TerraPower has that have this information to we think an appropriate degree. They did mention in their topical report that there's the potential for fuel -- or for materials other than U-10Zirc or HT-9 to be part of the fuel system.

We included a limitation and condition on there to essentially say if you are going to use these materials, you need to describe them a little bit more. But I will say that all the materials that they mentioned in the topical report are, you know, code-qualified materials that are generally used in the industry in these kind of applications. So they're not things that we're particularly concerned about.

One thing that's important for fuel is, you know, making sure that the end-state attributes from the manufacturing process are appropriately captured. And we thought that TerraPower did that well enough in the topical report.

So Goal 2 talks about margin to safety limits, and that's supported by design limits for normal operation of AOOs and then also for accidents. Part of that is defining the fuel performance envelope that you want to be working in. TerraPower provided those in a pin and assembly damage criteria that they flashed on the screen earlier, and that was consistent with the key mechanisms that we saw from the NUREG.

We haven't seen specific limits on any of those criteria yet, so that's something that we would need to better understand before the fuel is considered to be fully qualified. And I'll talk a little about how their operating envelope compares to the historical operating experience later.

And one thing here I grayed out evaluation model is available. They included a discussion on evaluation models. So we know that they have analytical methods to assess the fuel. We didn't really review them in this topical report.

We think that the methods that they have look like they have what they need, you know, in terms of geometry and fields and stuff like that to be able to model the fuel. But it's not like this topical report had a validation of those methods, because that data is still being collected. So that's something that we're going to have to deal with later on.

So we already talked about 211 on the previous slide. Then this talks about these, so these are the release limits under accident conditions. And I just wanted to mention here these two bullets that I wanted to highlight really relate to limitation and condition 5, and that's the specifying the retention and release requirements.

TerraPower said that those -- that was going to be done in the mechanistic source term topical report. So it was outside the scope of the fuel topical report review. And so we are actively reviewing that topical report.

And here I will talk about the safety limits for accidents and -- transients and accidents. So the fuel failure criteria that TerraPower came up with were we thought consistent with the key mechanisms that we identified for metallic fuel. The coolable geometry criteria was also consistent.

There was a discussion in the NUREG-CR about ejection of molten debris from the fuel. They -- we thought that those were precluded by having a limit against fuel melt. If you're not going to melt the fuel, there isn't really a mechanism to eject much debris from the fuel. There's a lot of run beyond cladding breach testing that shows that the fuel just kind of sits there and nothing really happens to it.

The negative reactivity insertion criteria we thought were adequate. But again, as with the discussion on AOOs and normal operation, we would still need to understand what the specific limits would be on these criteria.

I already touched on the evaluation model, so I'll just skip through this slide. Data, so there is, as TerraPower mentioned, a lot of historical data out there. We focused in our safety evaluation on the scope and applicability of the previous data and how that data supports TerraPower's acceptance criteria, which I will say they did a really good job of laying out in the topical report, you know, how --what testing supports each criterion.

The type I fuel design and geometry that they have is generally consistent with metallic fuel that was operated at EBR-II and FFTF. It's a little bit fatter than the EBR-II fuel, a little, just a tiny bit fatter than the FFTF fuel. You saw the length of the fuel columns is about the same as FFTF.

A significantly larger plenum, which is good for accommodating fission gas release. So those differences we think are either beneficial in terms of the plenum. And I think the fuel cladding is slightly thicker too. Or aren't expected to have much impact on the applicability of the historical data. They're small deltas.

The fuel operating parameters were also generally consistent with the past operating experience. Some of those parameters are at or maybe slightly beyond the historic database. But those deltas we think are small, and they're not expected to have a lot of effect. They would be addressed by the surveillance program or are covered by testing that TerraPower proposed to do.

And our overall conclusion is that the new data collection, so basically the historical data is generally applicable. Where there are gaps, the new data collection that TerraPower proposed is appropriate to fill those gaps.

Shifting gears a little bit to talk about the control assembly, so they're boron carbide pellets in a plenum that looks a little, or in a fuel -- control rod, not a fuel rod, that looks a little bit more sort of like an LWR rod, where it's got a plenum with a hold-down spring.

But they also are clad in HT-9 with an HT-9 wire wrap, like the fuel rods are. And they're in that sort of tight, hexagonal arrangement.

The one thing that is important about the control rod design that isn't necessarily super obvious from the discussions that we've had already is that each control assembly occupies its own space in the core with its own duct. There is then a control rod duct inside that duct that moves up and down.

And so they, as we talked about during the previous meeting, you know, there's primary and secondary control assemblies to try to meet that PDC-26 criterion. The differences are really the number of absorber pins and the dimensions of the control assembly.

And then I just have I think a single slide on qualification of the control assemblies. But you know, sort of boiling down the NUREG-2246 criteria aren't exactly applicable to a control assembly because it has different safety functions. But you can kind of think of analogous criteria, at least at a high level.

So you know, are there are appropriate controls on manufacturing? Yeah, we looked at what they provided in the topical report and what they referenced in terms of design documents. We got to look at those in an audit. We think that they've appropriately specified what the manufacturing looks like to, at least to the degree that it needs to be.

The design criteria we thought were all appropriate to make sure that the control rods could fill -- fulfill their safety function.

For evaluation model, kind of similar story as with the fuel rods where the codes we think have the ability to do what they need to do, but there's some validation that still needs to happen. And essentially I think that, as was mentioned during the TerraPower's meeting, it's the same codes, but they wanted -- they added boron carbide models.

For data, there is some historical data from past operating fast reactors for different control rod performance that TerraPower was able to draw on. I would say there's no exact one-to-one match for control rods in terms of like materials. And but there are -- there are some that use different combinations.

So we thought that the data that TerraPower was able to assemble looked like it covered the spectrum well enough. And they have planned testing again to fill gaps in this historical database.

Talked briefly about fuel surveillance and LDAs and LTAs, TerraPower touched on this. There’s a notional surveillance plan for the first several cycles of irradiation in the topical report. The LDAs and LTAs are designed with removable pins to facilitate close irradiation examination.

There’s significant precedent for a program like that, LDAs and LTAs, based on the operating fleet. We do want to see eventually more detail on how those leak demonstration, leak test assemblies will be evaluated.

To the point that you brought up, you know, the removable pins won’t have wire wrap. So how does that affect the performance of those pins and how do you evaluate it? That’s something that wasn’t necessarily clear from the topical report. So that’s something we’re going to dig into as we go forward.

Limitations and conditions. So the first one really is, you know, this is a good plan. But that does -- it's a plan and you still need to execute it. The second one is the point that I touched on about use of materials other than U-10Zirc and HT-9 in fuel. If you are going to use those, we need to talk more about it.

The topical report, sorry, the third criterion here relates to the relationship between the fuel design limits and the SRDLs. Essentially this is good fuel design limits that provides good context for evaluating the SRDLs. But you have to actually evaluate the SRDLs for like stochastic failures of fuel pins or whatever.

For number 4, and I can talk more about this if we want to have a closed session, but I did cover it during the subcommittee meeting. There were some documents that TerraPower referred to in the topical report for helping develop their design criteria that hadn't yet been the subject of NRC reviews. So we just wanted to point that, that criterion there or that L&C there.

And limitation 5 really relates to the retention of radionuclides. And we think that it's okay to push that off to a different topical report, we just wanted to put this limitation to make it clear what the scope of this topical is.

And our overall conclusion is that the topical report was acceptable and provided an overall acceptable approach for qualifying fuel and control assemblies.

And the one thing I will say, and this did come up last time, part of that is the monitoring and surveillance program we think is a really important part of the overall fuel qualification effort.

CHAIR KIRCHNER: Go back to No. 5, please.

So you would expect the actual performance would be in the mechanistic source term report?

MR. ANZALONE: So we left it open. I don't think we said this has to be in the mechanistic source term. But I think TerraPower has said that it's covered by the mechanistic source term topical report.

CHAIR KIRCHNER: Just always a little bit on guard, so to speak, when you have statements like are expected to remain within the fuel, etc. So this implies that they're going to demonstrate that or make the case somewhere else.

MR. ANZALONE: Yeah, in their evaluation of the source term, they would have to justify whatever is happening to the radionuclides. If they're crediting retention, say, like I think we have a reasonable expectation that a lot of radionuclide, especially solid fission products are going to be retained within the fuel matrix. So that's just based on, you know, the data that's out there.

But if TerraPower wants to credit that in their mechanistic source term analysis, that's something that they're going to have to talk about at that point. That's what this limitation --

CHAIR KIRCHNER: Yeah.

MEMBER PETTI: It doesn't imply that there isn't any release from cladding breach. Some fission products are coming out into the sodium, sure.

MR. ANZALONE: Yeah.

MEMBER PETTI: This data. Yeah, but there's a lot of other fission products --

MR. ANZALONE: Exactly.

CHAIR KIRCHNER: Steve, you have your hand up.

DR. SCHULTZ: Yes, thank you. Reed, you mentioned as you described the -- in particular the methodologies that are being used to evaluate the control rod performance, control element performance.

Do you -- can you expand on that, what you're looking for in terms of what additional work needs to be done there and when we can expect that information to come forward? Is it benchmarking, or something more than that?

MR. ANZALONE: So that's a good question. I think in general, like we -- I'm comfortable with the state of things as far as having like preliminary analyses to support the PSAR. I think we would want the sort of more full qualification to be done before the operating license, by the operating license.

I don't know if that answers your question, though.

DR. SCHULTZ: Well, you specifically mentioned that the methodologies would need additional attention. And is that what you're referring to there, that --

MR. ANZALONE: Yeah, yeah, that we would -- that we would need to have some way of validating, right, that. So say, you know, you -- there's going to be a pressurization of the control rods as you, you know, burn up the boron, for lack of a better word. So you would want to be able to make sure that those aren't going to break open and spill out all of their poison.

So we would need to be able to see eventually an evaluation of that and have some confidence that the models were validated for that.

MEMBER PETTI: Also I would think the dimensional change --

MR. ANZALONE: Yep.

MEMBER PETTI: Because of this in the past some either fuel or control assemblies stick and --

DR. SCHULTZ: Yes.

MR. ANZALONE: Yeah.

MEMBER PETTI: There's those issues that --

MR. ANZALONE: No, there are a lot -- there are a lot of different issues, yeah, absolutely. I was just giving that as an example.

DR. SCHULTZ: Thank you, that's helpful. Appreciate it.

MR. ANZALONE: But yeah, I think the big one is like dimensional change. And you can think about that either at like a pin level, right, you have swelling of the pin, and then maybe that stops there being appropriate cooling of the adjacent pins in the, you know. They have essentially subchannels too, like the fuel does. Or, at the assembly level you get deformation that stops it from being able to insert.

So that's definitely something that we want to pay attention to, because we think it's -- that's the key thing that drives the control rod insertability, is the assembly-level deformation.

DR. SCHULTZ: We talked in detail about the fuel, the fuel assembly qualification, fuel rod qualification program that's been proposed. Are you comfortable with what has been proposed with respect control rod?

MR. ANZALONE: Yeah. And I just didn't talk about it in as much detail in this presentation. I would say because the criteria are different, it's -- they're a little less tight because of the nature of control rods and their design function. But I would say that there's basically just as much in the topical report about control assembly qualification as there is fuel.

DR. SCHULTZ: Good, thank you.

MEMBER PETTI: So we, I know we talked about this in subcommittee. The whole qualification runs through all these codes. A heck of a lot of computer codes need a lot of data validation. And that always makes me a little bit nervous.

MR. ANZALONE: Yeah.

MEMBER PETTI: What I'm hoping is that the margin that, from an engineering gut feel that you have when you look at these designs, you look at performance, that that can translate through those codes to give you the analytical margins that you need when you have to go 95% confidence and stack up all these (audio interference).

That's the one thing that it's hard to see, the report doesn't really get into that at all. But it's the one thing that I worry about that when you get -- you don't know until you get --

MR. ANZALONE: Well, so it's on our minds too. I will say that that is one of the things that we're focused on looking at. Not this specific -- I mean, it's a through line for this topical report, right.

But it's as we're looking at like their design basis accident analysis methodologies, you know, we're thinking about what are their criteria that are in there for fuel failure and how are they actually evaluating that. So you're going to see more of that as we come through the reviews.

MEMBER PETTI: So you know, I went back and read the SER on PRISM, and there's an appendix that they did, the staff had some of our labs do calculations. And frankly the results, we're talking 1990s, really quite good comparing GE and lab tools. Granted, on reactivity they used the same reactivity coefficients, but still the results were really,

really quite good.

But it never got into the details of fuel model.

MR. ANZALONE: Sure.

MEMBER PETTI: And, you know, there's always such a mixture of empiricism and semi-empiricism. And now there's better models, but it remains to be seen that the sharper pencil gets you the answer you want.

This has always been one of my concerns about these really cool advanced models. Hopefully they verify your engineering judgment. But that all of that effort gets you margin and all of that in the end you stack it all together.

MR. ANZALONE: Yeah, totally agree. I'm 100% aligned on that.

MEMBER PETTI: Good.

MR. ANZALONE: And you know, one thing I'll say is that we're talking to the Office of Research about ways in which they can support us with doing confirmatory analyses and stuff like that, as was done for the PRISM review.

Some of those I think would use our codes, some of those might use, depending on where things go, you know, the NEAMS codes like BISON or what have you for fuel performance, so.

MEMBER PETTI: There's some good publications already out there on BISON, the amount of fuel, that I found really helpful.

MR. ANZALONE: And I think one thing that I am trying to be cognizant of when we have those conversations is, you know, to what degree are -- because this is something that was brought up in that NUREG-CR, to what degree are those mechanistic models actually well-validated and is there the data to support them. I think that it kind of remains to be seen a little bit.

But TerraPower, I think that their approach that they're taking is solid, so, not too concerned with their modeling approach here.

MEMBER ROBERTS: If there's no more questions from members or consultants? I guess it's time now to go out for public comments.

If there's any members of the public who'd like to make a comment, please go ahead and unmute yourself, state your name and affiliation if there is one, and then state your comment, please.

Hearing none, guess I'll turn the meeting back over to Chair Kirchner.

CHAIR KIRCHNER: Thank you, Tom and Dave.

Thank you to the presenters, both staff and Applicant, thank you.

And at this point, we're actually a little ahead of schedule for quite a change. And so we've set aside a period now to have committee deliberation on what we heard on both of these topical reports. And then we can proceed at this point with our letter writing.

So, Jose, would you like to make a comment?

MEMBER MARCH-LEUBA: Court reporter, is he needed the rest of the week?

CHAIR KIRCHNER: Let me confer with Larry. Do, at this point do we need the court reporter further?

MR. BURKHART: I think we're going into deliberation and letter writing. We can let the court reporter loose.

CHAIR KIRCHNER: Looking at the schedule for today and tomorrow, and --

MR. BURKHART: It is all we have left, yes.

CHAIR KIRCHNER: We P&P tomorrow, and so we --

MR. BURKHART: Correct.

CHAIR KIRCHNER: Don't normally record that, correct?

MR. BURKHART: We don't, no.

CHAIR KIRCHNER: With that, okay.

For the court reporter, thank you. I don't believe that we'll need your services for the rest of this meeting and this week.

(Whereupon, the above-entitled matter went off the record at 11:11 a.m.)

Staff Review of NATD-FQL-PLAN-0004, “Fuel and Control Assembly Qualification”

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Agenda

- Topical report (TR) purpose and review strategy
- Regulatory requirements and guidance
- Overview of NUREG/CR-7305
- Safety evaluation (SE) overview
  - Fuel assembly design and qualification
  - Control assembly design and qualification
  - Surveillance, lead demonstration / lead test assemblies (LDAs/LTAs)
  - Limitations and conditions
- Conclusions

TR Purpose

- Provides plan to qualify Natrium Type 1 fuel (uranium-zirconium alloy in HT9 cladding) and Natrium control assemblies
- Requests NRC review and approval of the following:
  - Acceptance criteria are adequate to support fuel qualification
  - Identified key manufacturing parameters are adequate to support fuel qualification
  - Evaluation methods and models are adequate to support fuel qualification
  - Use of legacy data and planned testing are adequate to provide necessary information for qualification of the fuel
  - Planned use of pins outside the performance envelope of the bulk of the core or that advanced design features are acceptable
- Presents select fuel qualification results and ongoing and planned qualification activities

TR Review Strategy

- Review scope and adequacy of fuel qualification plan in the context of NUREG-2246, “Fuel Qualification for Advanced Reactors” (ML22063A131)
- Review technical details of fuel and qualification efforts against information in NUREG/CR-7305, “Metal Fuel Qualification: Fuel Assessment Using NRC NUREG-2246, ‘Fuel Qualification for Advanced Reactors” (ML23214A065)

Regulatory Requirements

- 10 CFR 50.43(e)
  - Requires safety features to be supported by analysis, testing, operating experience, or a combination thereof.
  - Requires sufficient data exists to assess analytical tools
- 10 CFR 50.34
  - Requires applicants to evaluate a postulated fission product release from the core into containment
  - Requires principal design criteria (PDCs) to be submitted

NUREG-2246: Fuel qualification provides a means to identify safety criteria for the fuel, which then are used to establish performance criteria for facility structures, systems, and components (SSCs). Facility safety is then addressed by description and analyses of these SSCs.

Guidance

- NUREG-2246, “Fuel Qualification for Advanced Reactors”
  - Provides general guidance on fuel qualification for non-light water reactors (non-LWRs) in the form of a Fuel Qualification Assessment Framework (FQAF)
- NUREG/CR-7305 “Metal Fuel Qualification: Fuel Assessment Using NRC NUREG-2246, ‘Fuel Qualification for Advanced Reactors”
  - Provides a generic response to NUREG-2246 for a uranium-zirconium metal fuel system, including
    - Identification of an operating envelope and key behaviors/phenomena
    - Review of available data
    - Discussion of current state of fuel performance modeling

NUREG/CR-7305 – Key Conclusions

- For fuel with geometry and operating conditions consistent with previous operating experience, “life-limiting and safety-related fuel behaviors are well known and predictable” up to 10 atom-% burnup
  - Fuel constituent redistribution is captured in data
  - FCCI is life-limiting phenomenon
  - FCMI is not a concern
- Additional transient data would help to establish safety margins
- More work is needed to qualify mechanistic models

Natrium Fuel Assembly Design

- Very similar to EBR-II fuel and metallic fuel operated at the Fast Flux Test Facility (FFTF)
- Pin characteristics:
  - Metallic uranium alloyed with 10 weight-% zirconium (U-10Zr)
  - Peak enrichment < 20%
  - 75% smear density
  - Sodium bond
  - HT9 cladding
  - Axial shield slug
  - Large plenum
  - HT9 wire wrap
- Assembly characteristics
  - Pins arranged in tight triangular pitch in hexagonal bundle
  - Hexagonal duct, inlet nozzle, handling socket
- Limited free bow core restraint system

NUREG-2246 FQAF

- G1. Fuel is manufactured in accordance with a specification
  - G1.1 Key dimensions and tolerances of fuel components are specified
  - G1.2 Key constituents are specified with allowance for impurities
    - TR includes/refers to adequate design information
    - L&C #2 covers use of materials other than U-10Zr/HT9
  - G1.3 End state attributes for materials within fuel components are specified or otherwise justified.
    - Adequate end state attributes provided

NUREG-2246 FQAF

- G2. Margin to safety limits can be demonstrated.
  - G2.1 Margin to design limits can be demonstrated under conditions of normal operation and AOOs.
    - G2.1.1 Fuel performance envelope is defined
      - Pin and assembly damage criteria consistent with key mechanisms from NUREG/CR-7305
      - Specific limits must be provided before fuel is considered qualified (L&C #1)
      - Comparison to fuel operating experience discussed later
    - G2.1.2 Evaluation model is available

NUREG-2246 FQAF

- G2. Margin to safety limits can be demonstrated.
  - G2.2 Margin to radionuclide release limits under accident conditions can be demonstrated.
    - G2.1.1 Fuel performance envelope is defined
    - G2.2.1 Radionuclide retention requirements are specified
      - Addressed in separate TR; L&C #5
    - G2.2.2 Criteria for barrier degradation and failure are suitably conservative
    - G2.2.3 Radionuclide retention and release from fuel matrix are modeled conservatively
      - Addressed in separate TR; L&C #5

NUREG-2246 FQAF

- G2. Margin to safety limits can be demonstrated.
  - G2.2 Margin to radionuclide release limits under accident conditions can be demonstrated.
    - G2.2.2 Criteria for barrier degradation and failure are suitably conservative
  - G2.3 Ability to achieve and maintain safe shutdown is assured.
    - G2.3.1 Coolable geometry is ensured
    - G2.3.2 Negative reactivity insertion can be demonstrated
- Fuel failure criteria consistent with key mechanisms from NUREG/CR-7305
- Coolable geometry criteria are consistent with NUREG/CR-7305, except molten debris ejection which is precluded by preventing fuel melt
- Negative reactivity insertion criteria are adequate
- Specific limits must be provided before fuel is considered qualified (L&C #1)

Evaluation Models

- Separate EM assessment framework in NUREG-2246
- TR does not contain detailed information on fuel performance models, staff did not fully assess against NUREG-2246 framework
- Codes discussed in TR appear to provide the capabilities needed to support fuel qualification efforts
- Additional effort is needed to demonstrate that the proposed EMs contain all necessary material and physics models, verify the EMs, and validate them against experimental data.
  - EMs will be evaluated in future revision of this TR or in a separate TR specifically covering fuel performance

Data

- Because evaluation of historical data and data collection is ongoing, focus in SE is on scope and applicability of historical data, how data supports TerraPower's acceptance criteria, and plans for future testing
- Type 1 fuel design geometry is generally consistent with metallic fuel operated at EBR-II and FFTF
  - Differences either beneficial or not expected to have much impact on applicability of historical data
- Fuel operating parameters also generally consistent with EBR-II/FFTF
  - Some parameters are at or slightly beyond historical database
  - Deltas are small, and are not expected to have much effect, will be addressed by TerraPower's planned surveillance program, and/or will be the subject of proposed testing discussed in the TR
- New data collection appropriate to fill gaps

Control Assembly Designs

- Pin characteristics
  - Natural boron carbide pellets
  - Plenum with spring
  - HT9 cladding
  - HT9 wire wrap
- Assembly characteristics
  - Triangular pitch in hexagonal lattice
  - Upper guide plate with coupling head
  - Control rod duct that moves up and down inside control assembly duct
- Primary/secondary control assembly differences
  - Number of absorber pins
  - Dimensions, including space between inner control rod duct and control assembly duct

Control Assembly Qualification

- Manufacturing
  - Control assembly manufacturing appropriately specified
- Design criteria
  - Damage, failure, and insertability criteria adequate to ensure control rods can fulfil their safety function
- Evaluation model
  - Same codes as fuel assemblies with changes for control assemblies
  - Codes appear capable but more work is needed
- Data
  - Historical data from EBR-II, FFTF, Joyo
  - Planned testing to fill gaps in historical data

Fuel Surveillance, LDAs, and LTAs

- TR presents notional surveillance plan for first several cycles
- LDAs and LTAs designed with removable pins to facilitate post-irradiation examination (PIE)
- Significant precedent for LDA/LTA program based on operating fleet
- Additional detail required on how LDAs/LTAs will be evaluated and how uncertainties in performance will be captured in analyses

L&Cs

- 1. This TR represents an acceptable approach for qualifying Natrium Type 1 fuel and control assemblies for use in a reactor but does not in and of itself demonstrate that the fuel and control assemblies are qualified. Additional activities, including those discussed in the NRC staff's SE, must be completed to execute this plan and appropriately justify that the fuel and control assemblies are qualified.
- 2. This TR addresses the material properties and performance of U-10Zr and HT9 in fuel. If other materials are used in the fuel system in licensing applications, the applicant or licensee must demonstrate that they are manufactured according to standard specifications and used consistent with their qualification under relevant NRC-accepted codes and standards, or otherwise appropriately justified.

L&Cs

- 3. This TR does not provide a means for demonstrating that proposed SARRDLs are satisfied during normal operations and AOOs for the Natrium plant. The role of the fuel acceptance criteria is to demonstrate that the fuel system is not damaged as a result of normal operations and AOOs; if these criteria are satisfied, then the fuel system need not be further assessed against the SARRDLs. However, the SARRDLs must still be evaluated against other sources of radionuclides, including circulating radionuclides resulting from an appropriate number of random fuel failures.
- 4. The [[ ]] have not been subject to previous NRC review or approval. If they are to be used to develop design criteria and associated limits that support fuel assembly acceptance criteria, these design criteria and associated limits must be appropriately justified.

L&Cs

- 5. This TR does not address the extent to which the fuel system is expected to retain radionuclides following a cladding breach. If an applicant or licensee wishes to qualify Natrium Type 1 fuel with an expectation that radionuclides are expected to remain within the fuel following a cladding breach, models for fuel system radionuclide retention and release must be proposed and appropriately justified by comparison to experimental data.

Conclusions

TR is acceptable for referencing in future licensing submittals, subject to limitations and conditions.

- The NRC staff determined that the TR provides an acceptable approach for qualifying fuel and control assemblies for the Natrium reactor based on
  - (1) the inclusion of sufficient information to demonstrate that fuel and control assemblies are manufactured in a process that provides adequate control over key parameters,
  - (2) the identification of appropriate safety criteria for both fuel and control assemblies,
  - (3) the development and justification of a significant applicable historical test database,
  - (4) the development of a test plan that appropriately fills gaps in the historical test database, and
  - (5) a robust fuel monitoring program, subject to the limitations and conditions discussed above. Accordingly, the NRC staff concludes that the qualification plan provided in the TR can be used to support compliance with 10 CFR 50.43(e) and proposed Natrium PDCs.

Abbreviations

ACCI – Absorber-cladding chemical interaction

AOO – Anticipated operational occurrence

CFR – *Code of Federal Regulations*

EBR-II – Experimental Breeder Reactor-II

EM – Evaluation model

FCCI – Fuel-cladding chemical interaction

FCMI – Fuel-cladding mechanical interaction

FFTF – Fast flux test facility

FQAF – Fuel qualification assessment framework

LDA – Lead demonstration assembly

LTA – Lead test assembly

Non-LWR – Non-Light Water Reactor

PDC – Principal design criterion

PIE – Post-irradiation examination

Pu – Plutonium

RAC – Regulatory Acceptance Criteria

SARRDL – Specified acceptable radionuclide release design limit

SE – Safety evaluation

SSC – Structure, system, or component

TR – Topical report

TRISO – Tri-structural Isotropic

U – Uranium

Zr - Zirconium

Review Chronology

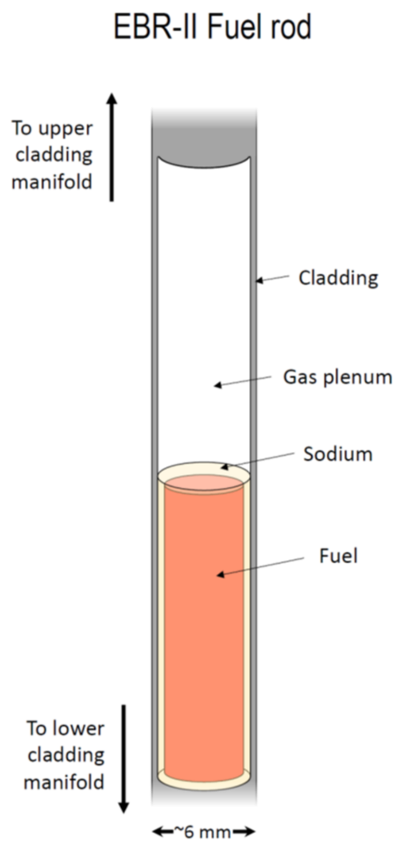
- January 25, 2023: Submittal of TR “Fuel and Control Assembly Qualification Plan,” Revision 0 (ML23025A409)
- March 21, 2023: Pre-Application Public Meeting (ML23157A332)
- March 31, 2023: TR accepted for review by the NRC staff (ML23086C087)
- April 18, 2023: Submittal of correction to TerraPower Fuel and Control Assembly Qualification Topical Report (ML23109A099)
- June, July, and August 2023: Audit Conducted (ML24043A155)
- March 20, 2024: Draft SE Issued (ML24079A118)

NUREG-2246 FQAF

G. Fuel is qualified for use.	G1. Fuel is manufactured in accordance with a specification.	G1.1 Key dimensions and tolerances of fuel components are specified.	
		G1.2 Key constituents are specified with allowance for impurities.	
		G1.3 End state attributes for materials within fuel components are specified or otherwise justified.	
	G2. Margin to safety limits can be demonstrated.	G2.1 Margin to design limits can be demonstrated under conditions of normal operation and AOOs.	G2.1.1 Fuel performance envelope is defined
			G2.1.2 Evaluation model is available
		G2.2 Margin to radionuclide release limits under accident conditions can be demonstrated.	G2.1.1 Fuel performance envelope is defined
			G2.2.1 Radionuclide retention requirements are specified
			G2.2.2 Criteria for barrier degradation and failure are suitably conservative
			G2.2.3 Radionuclide retention and release from fuel matrix are modeled conservatively
		G2.3 Ability to achieve and maintain safe shutdown is assured.	G2.3.1 Coolable geometry is ensured
			G2.3.2 Negative reactivity insertion can be demonstrated

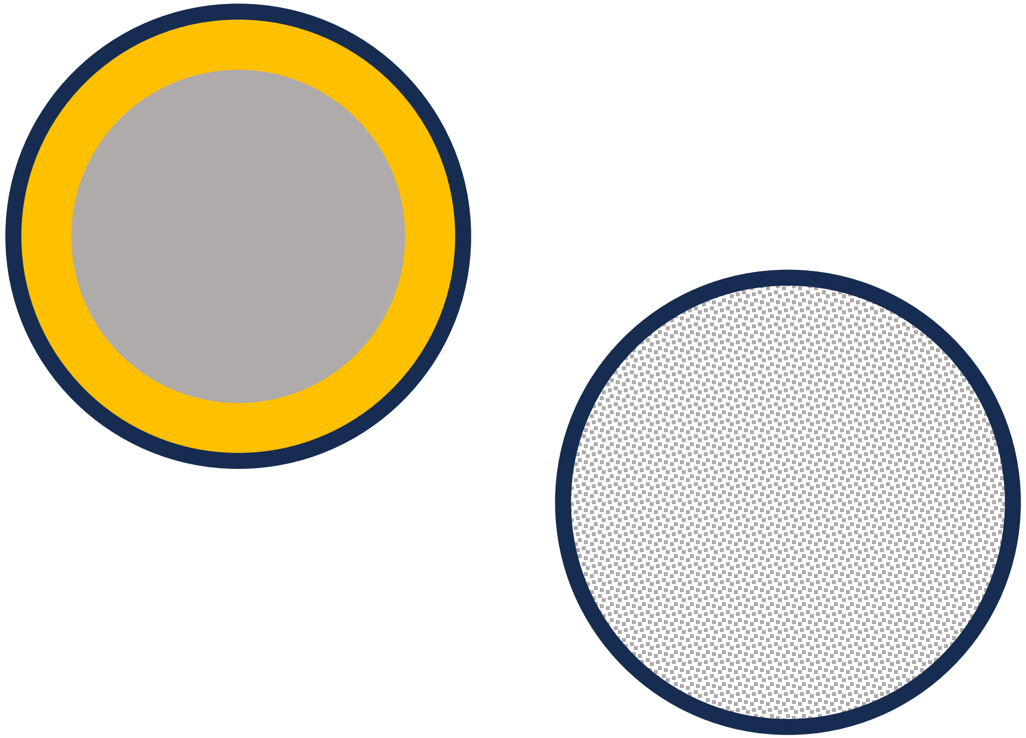
NUREG/CR-7305 Design Parameters

- Uranium-10 weight% zirconium alloy fuel
- 75% smear density
- 1.4 plenum to fuel volume ratio
- Sodium bond
- HT9 cladding
- Fuel dimensions from Experimental Breeder Reactor-II (EBR-II)

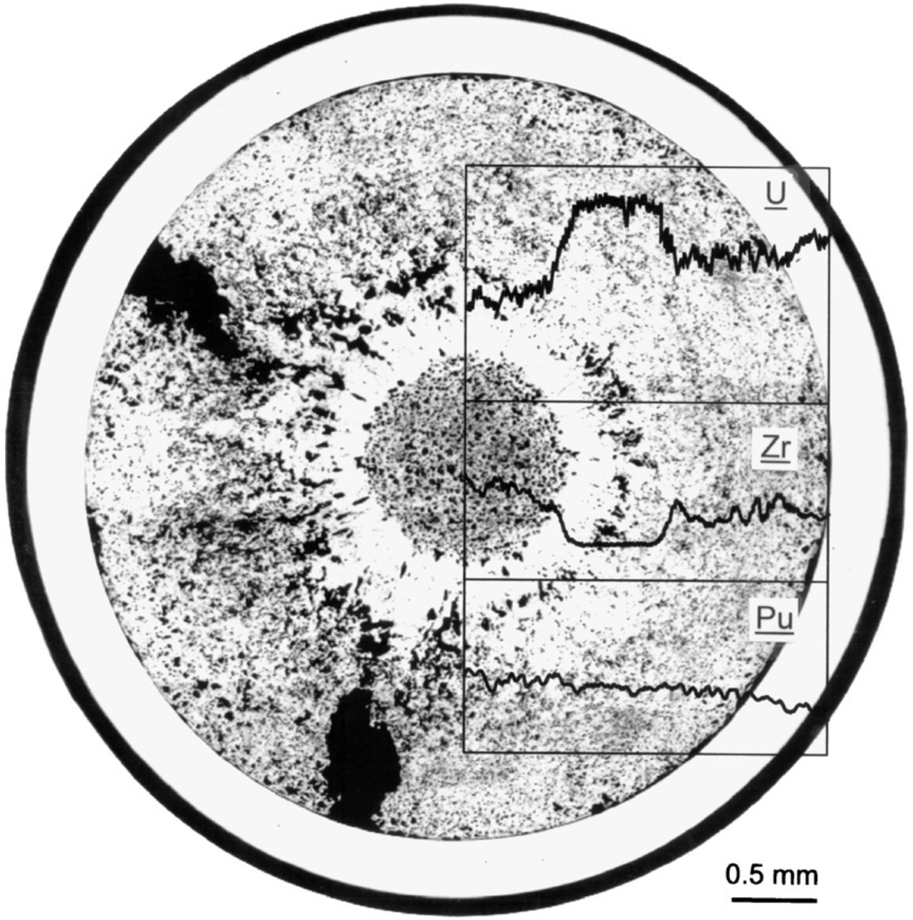


NUREG/CR-7305 – Fuel Geometric Evolution

- Fuel swells axially and radially until cladding contact
- Porosity interconnects and gaseous fission products are released to the plenum
- Solid fission product build up
- At >10 atom% burnup, fission gas flow through pores becomes constrained and fuel begins to swell again



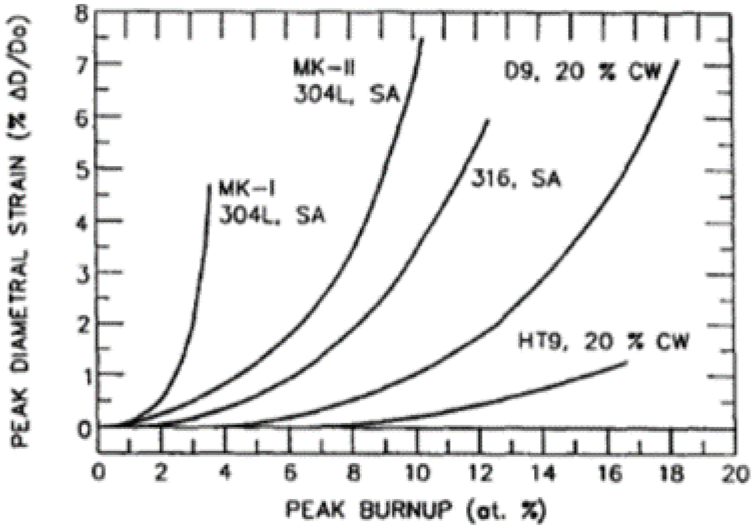
NUREG/CR-7305 – Fuel Constituent Redistribution



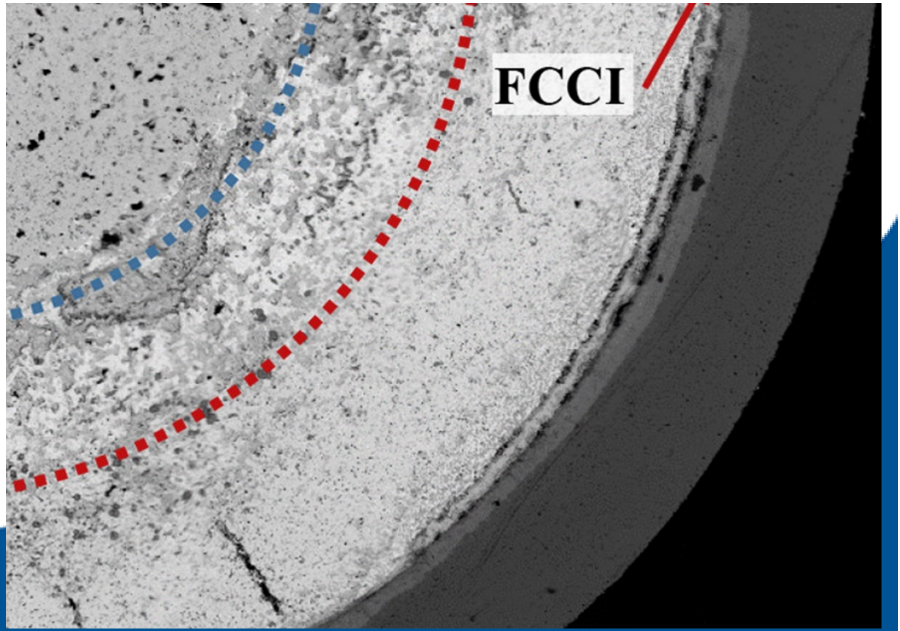
- Thermal gradient in fuel drives redistribution of U and Zr in fuel
- Higher operating temperatures and linear heat rates drive more redistribution
- Potentially affects fuel properties, local power density
- Accounted for in experimental data below 10% burnup

Kim, Yeon Soo, S. L. Hayes, G. L. Hofman, and A. M. Yacout. "Modeling of constituent redistribution in U–Pu–Zr metallic fuel." *Journal of Nuclear Materials* 359, no. 1-2 (2006): 17-28.

NUREG/CR-7305 – Cladding Integrity/Barrier Degradation



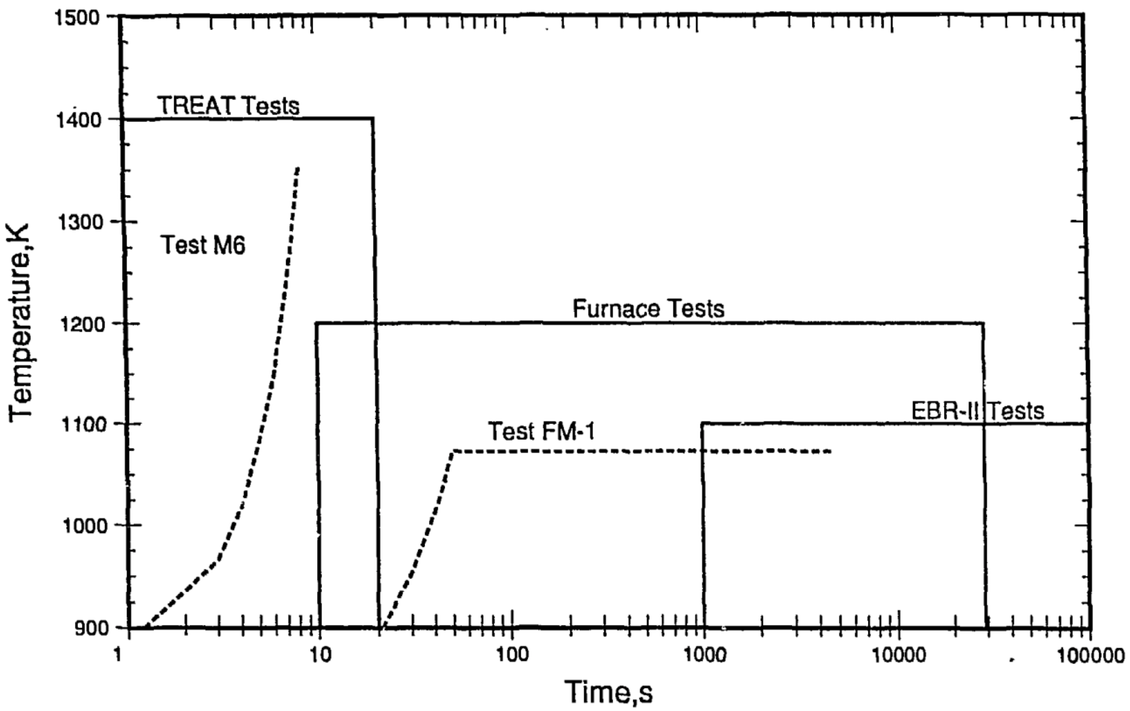
- Fuel-cladding mechanical interaction (FCMI) not a concern below 10% burnup for fuels with 75% smear density
- Fission gas release not a concern with appropriately sized plena
- Fuel-cladding chemical interaction (FCCI) is primary source of cladding degradation and fuel failure
  - Thins cladding due to formation of low-melting point eutectics at fuel-cladding interface
    - U-Fe but also contributed to by lanthanides, which tend to migrate down thermal gradient
  - Measurable thinning at ~725°C, NUREG/CR recommends steady-state limit of 650°C



NUREG/CR-7305 – Fuel Properties

- Porosity and redistribution evolution affect properties
- Significant margin to solidus temperature (>1100°C); bulk fuel melting is not a concern and FCCI region provides limit for fuel temperature
- Limited thermal conductivity data but favorable compared to UO_2
- Limited irradiated mechanical properties but below 10% burnup, empirical models adequately predict fuel swelling and cladding strains

NUREG/CR-7305 – Transients



- Transient testing (in-pile and out-of-pile) has been done and identified FCCI to be the primary failure mode
- Additional transient testing is needed to characterize operating envelope

G1.1 & G1.2 – Key Dimensions & Constituents

- TR refers to design drawings and materials specifications
- TR also includes details on HT9 and U-10Zr composition
- Staff audited referenced documents and found that they contained appropriate information.
- Use of materials other than U-10Zr and HT9 not clear – Limitation and Condition (L&C) 2

G1.3 – End-State Attributes

- NUREG/CR-7305 provides details on manufacturing process and important end-state attributes for U-10Zr, summarized as:
  - Injection molding with controls on formation of oxides and fuel density
  - Limited voids in sodium bond and appropriate amount of sodium
  - Fuel rod plenum sized appropriately
- Manufacturing process discussed at high level in TR, with references to specifications, including fabrication process
- Consistent with end-state attributes discussed in NUREG/CR

G2.1.1 – Fuel Performance Envelope

- TerraPower developed Regulatory Acceptance Criteria (RAC) for different mechanisms to provide an envelope in which fuel damage can be precluded
- “Damage”: Fuel has not failed but may have reduction in functional capability (i.e., outside of safety analysis assumptions)

G2.1.1 – Fuel Performance Envelope

- Pin Damage Criteria
  - Stress, strain, loading
  - Fatigue
  - Fretting wear
  - Erosion and corrosion
  - Cladding damage due to FCCI
  - Dimensional changes (rod bowing or swelling)
  - Pin internal pressure
  - Fuel and cladding temperatures

- Assembly Damage Criteria
  - Stress, strain, loading
  - Fatigue
  - Fretting wear
  - Erosion and corrosion
  - Dimensional changes (duct bowing and dilation)
  - Hydraulic loads exceeding hold-down
  - Assembly component temperatures

G2.1.1 – Fuel Performance Envelope

- Damage mechanisms presented are consistent with key phenomena and properties from NUREG/CR-7305
- Staff did not evaluate specific limits to prevent damage, which are expected to be under development as part of fuel qualification plan (L&C 1)
- Operating envelope and comparison to historical data is discussed in more detail in experimental data assessment framework

G2.1.1 – Fuel Performance Envelope (Accidents)

- TerraPower developed separate RAC for accidents; these are assessed under separate goals for barrier failure, radionuclide retention and release, coolable geometry, and negative reactivity insertion

G2.2.1 – Radionuclide retention requirements
G2.2.3 – Radionuclide release modeling

- Addressed in separate TR (source term methodology)
- L&C 5

G2.2.2 – Barrier Degradation & Failure Criteria

- Barrier degradation criteria covered under G2.1.1
- Pin failure criteria include:
  - Cladding and slug overheating
    - For gross melting but also rapid eutectic penetration
  - Cladding deformation due to mechanical loads
  - Fuel system mechanical fracturing from externally applied forces
  - Cladding wastage (including wear, erosion, corrosion, FCCI, eutectics)
- Consistent with discussion in NUREG/CR-7305
- Future work to establish appropriate limits (L&C 1)
- Supporting data discussed in separate framework

G2.3.1 – Coolable Geometry

- TerraPower developed separate RAC related to coolable geometry:
  - Stress and strain limits to ensure coolability
  - Cladding and fuel temperatures below melting point
  - Coolability evaluations must include cladding ballooning
  - Structural deformation of fuel assemblies cannot prevent core cooling
  - Hydraulic loads cannot unseat assemblies such that flow is reduced enough to prevent assembly cooling
- Generally consistent with NUREG/CR-7305, except debris ejected from failed fuel assemblies not explicitly addressed
  - Based on historical data, preventing fuel melt precludes this issue

G2.3.2 – Negative Reactivity Insertion

- Negative reactivity insertion sensitive to control assembly distortion, unseating of control assemblies
- TerraPower developed separate RAC related to reactivity insertion:
  - Structural deformation of control assemblies will not prevent the ability to insert control rods during accidents
  - Hydraulic loads will not unseat control assemblies in a way that prevent insertion during accidents
- Other RAC also help ensure insertability:
  - Fuel and control assembly distortion
  - Fuel and absorber pin internal pressure
  - Hydraulic loading on control assemblies
  - Mechanical/neutronic design of control assemblies
- Criteria address possible mechanisms

NRC Staff Review of the Topical Report "Principal Design Criteria for the Natrium Advanced Reactor," Revision 1

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Reed Anzalone, Senior Nuclear Engineer

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and
Non-Power Production and Utilization Facilities

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

Agenda

- Topical Report (TR) purpose and review strategy
- Regulatory requirements
- Natrium principal design criteria (PDC) overview
- Key topics from ACRS subcommittee meeting
  - Functional containment
  - Specified acceptable system radionuclide release design limits (SARRDLs)
  - Limitations and conditions

TR Purpose and Review Strategy

- Purpose of TR:
  - Describe TerraPower's process for developing PDCs
  - Provide PDCs to address compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(3)(i) for Construction Permit (CP) applications
  - Describe rationale for meeting the intent of Natrium PDC 26, "Reactivity Control Systems"
- Review strategy
  - Review Natrium PDC conformance with Regulatory Guide (RG) 1.232; group and evaluate deviations, considering key design features
  - Identify interaction between RG 1.232 approach and Licensing Modernization Project (LMP)
  - Review PDC 26 rationale

Regulations

- 10 CFR 50.34(a)(3)(i) requires an applicant for a CP to include the PDCs for the facility in the preliminary safety evaluation report (PSAR)
- 10 CFR 50, Appendix A provides requirements on the scope and content of PDCs for non-light water reactors (non-LWRs):
  - “The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.”
  - “These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.”

Natrium PDC Overview

- TerraPower developed PDCs based on RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors" (ML17325A611)
- Most PDCs based on SFR-DC (Appendix B of RG 1.232)
  - 1-12, 14, 15, 17-19, 21-24, 26, 28-37, 44-46, 60-64, and 70-79
- Some PDCs based on MHTGR-DC (Appendix C of RG 1.232)
  - 13, 16, 20, 25, 80, 81, and 82
  - Used to implement functional containment or reflect use of SARRDLs
- Most PDCs are modified from the RG 1.232 DC
- No DC for 38-43, 50-57 due to use of functional containment

General Changes to PDCs

A. Use of the term “safety-significant”
B. Use of graded approach to coolant boundary quality
C. Use of specified acceptable system radionuclide release design limit
D. Use of functional containment concept
E. Minor generic changes

Functional Containment Overview (1)

- RG 1.232, Appendix C, MHTGR-DC 16 (ML17325A611):
  - "The term 'functional containment' is applicable to advanced non-LWRs without a pressure retaining containment structure. A functional containment can be defined as 'a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions.'"

Functional Containment Overview (2)

- SECY-18-0096 (ML18115A157) documents approach to determining functional containment performance criteria
  - Technology-inclusive, risk-informed, performance-based
  - Methodology later developed into LMP
  - Developed in parallel with RG 1.232
- **SRM-SECY-18-0096 (ML18338A502) documents the Commission's approval of the NRC staff's approach to determining functional containment performance criteria for non-LWRs.**

Non-Applicability of Containment Criteria

- TerraPower did not adopt DC 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56, 57
- MHTGR-DC rationales note that these criteria are not applicable because there is not a “pressure containing reactor containment structure”
- Some relevant SFR-DC note that they would not be applicable if alternate approaches to containment were taken:
  - SFR-DC 38: “‘…as necessary…’ is meant to condition an SFR-DC 38 application to designs requiring heat removal for conventional containments that are found to require heat removal measures.”
    - SFR-DC 39 and 40 directly support 38
  - SFR-DC 50 references a containment structure; 51-57 support 50 and state they are applicable to designs employing containment structures.

Natrium Functional Containment Considerations

- SECY-18-0096 and associated SRM indicates that functional containment concept is acceptable for non-LWRs
- Staff's finding is that certain reactor attributes are necessary for functional containment approach to be viable for Natrium; actual functional containment performance remains to be demonstrated
- Use of LMP implies method used to demonstrate functional containment performance:
    - PRA and mechanistic source term analyses will be performed and must meet criteria (discussed in NEI 18-04, consistent with SECY-18-0096)
    - Analyses will explicitly consider uncertainties
    - Plant design will be evaluated for defense-in-depth adequacy per NEI 18-04

SARRDLs

- Initially identified in RG 1.232, Appendix C, MHTGR-DC 10 for TRISO
- SARRDLs are for normal operation and AOOs, are established so 10 CFR Part 20 limits are not exceeded
- SECY-18-0096, Enclosure 2 (ML18115A367):
  - "Defining SARRDLs for specific designs is intertwined with functional containment performance criteria and would be developed by reactor designers as part of the integrated approach described in this enclosure."

Natrium SARRDL Considerations

- Staff's view is that SARRDLs are appropriate to use with functional containment and are consistent with a performance-based evaluation of releases
  - TerraPower's fuel includes fuel design limits that can be used to help evaluate compliance with SARRDLs
  - SARRDLs can be a useful tool for looking at ex-vessel events
- Means of monitoring would need to be included as part of design
- TerraPower must still propose and evaluate SARRDLs
  - SARRDLs were discussed with TerraPower in a July 11, 2023, public meeting. Closed discussion included examples.

Scope and Applicability of PDCs

- Proposed PDCs are based on RG 1.232 (traditional framework) but applied to licensing under NEI 18-04 (risk-informed, performance-based framework)
- RG 1.253 provides guidance on scope of PDCs for LMP applications: "proposed PDC will need to address the functions provided by both SR and NSRST [non-safety related with special treatment] SSCs"
- Proposed limitation 2 addresses potential gaps
  - Will be addressed in CP application

Proposed Limitations and Conditions (L&Cs)

The NRC staff imposes the following L&Cs regarding the TR:

1. An applicant or licensee referencing this TR must propose a design that is substantially similar to the Natrium design as discussed in SE Section 1, or otherwise justify that any departures from these design features do not affect the conclusions of the TR and this SE.
2. The use of this TR is restricted to those applicants using the risk-informed, performance-based licensing process described in NEI 18-04, Revision 1, as endorsed by RG 1.233. Because the proposed PDCs may not fully address all performance requirements for SSCs defined as safety-significant under the NEI 18-04 process, applicants or licensees referencing this TR must augment the PDC in the TR with appropriate PDC for any SR or NSRST SSCs whose safety function relates to BDBEs, or NSRST SSCs needed for DID adequacy, or otherwise justify that the Natrium PDCs as described in the subject TR are adequate.

Conclusions

- TerraPower considered each of the design aspects presented in RG 1.232.
- TerraPower provided a sufficient set of PDCs for the Natrium design, subject to the L&Cs.
- The PDCs (subject to the L&Cs) establish the necessary design, fabrication, construction, testing, and performance DC for safety significant SSCs to provide reasonable assurance that the Natrium reactor could be operated without undue risk to the health and safety of the public.
- The TR is suitable for referencing in future licensing applications for the Natrium advanced reactor.

Abbreviations

ARDC – Advanced reactor design criteria

AOO – Anticipated operational occurrence

BDBE – Beyond design basis event

CFR – *Code of Federal Regulations*

CP – Construction permit

DANU - Division of Advanced Reactors and Non-Power Production and Utilization Facilities

DC – Design criterion

DBA – Design basis accident

DBE – Design basis event

GDC – General design criterion

L&C – Limitation and/or condition

LWR – Light water reactor

MHTGR – Modular high temperature gas reactor

NEI – Nuclear Energy Institute

NRR - Office of Nuclear Reactor Regulation

NSRST – Non-safety related with special treatment

NST – Non-safety related with no special treatment

PDC – Principal design criterion

PSAR – Preliminary safety evaluation report

QA – Quality assurance

RAC – Reactor air cooling system

RG – Regulatory guide

SAFDL – Specified acceptable fuel design limit

SARRDL – Specified acceptable system radionuclide release design limit

SFR – Sodium fast reactor

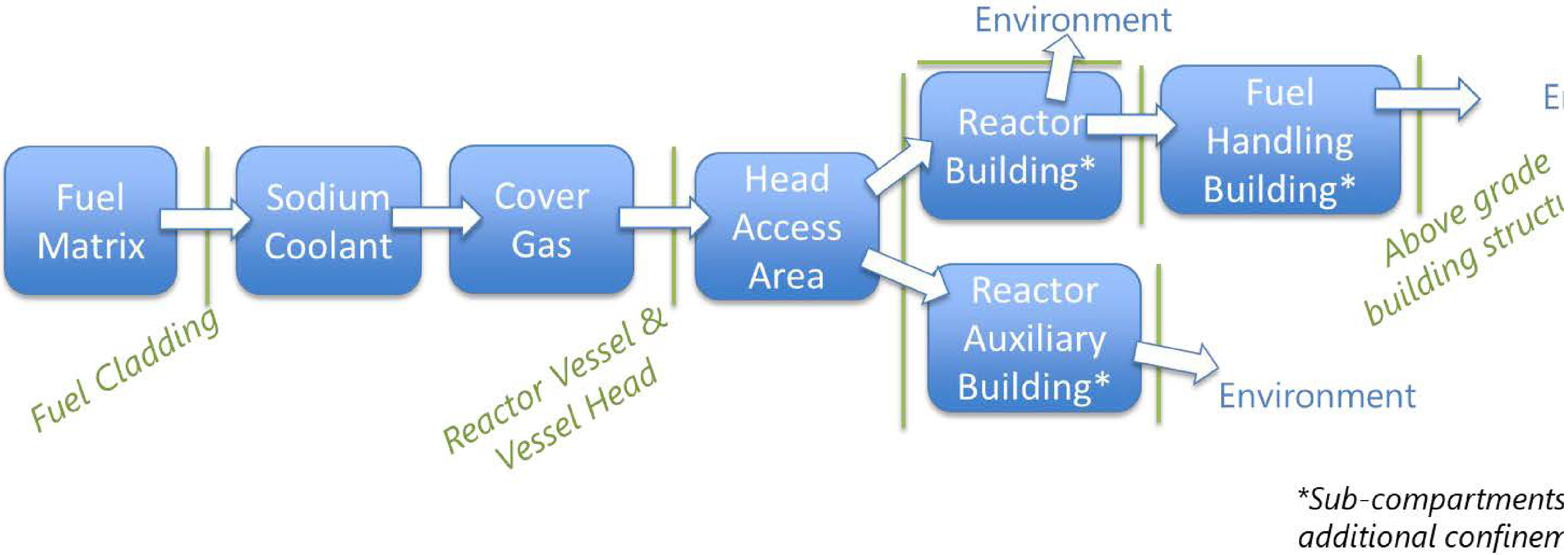
SSC – Structure, system, or component

SE – Safety evaluation

SR – Safety related

TR – Topical report

Natrium Functional Containment

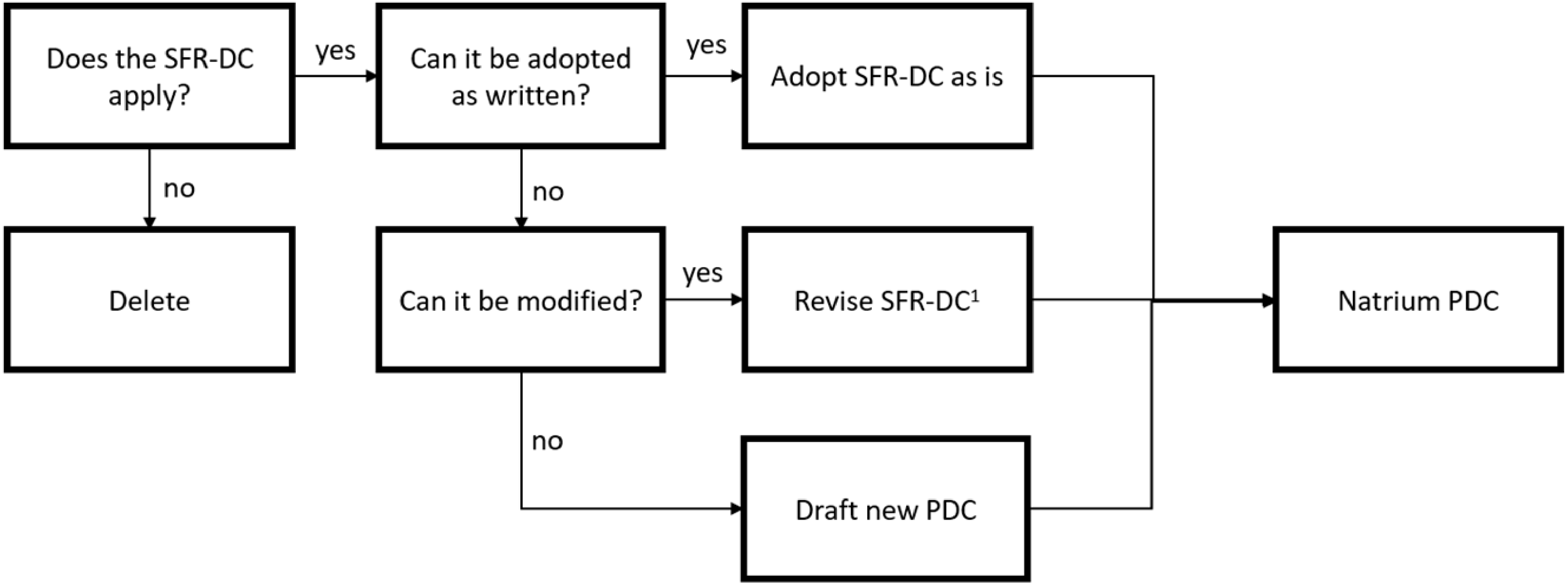


Natrium SSCs Associated with Functional Containment Strategy

- Metallic fuel matrix and cladding
- Reactor enclosure system, head access area, and primary coolant boundary
- Sodium processing system
- Sodium cover gas system
- Intermediate heat transport system
- Reactor building
- Reactor auxiliary building
- Water pool fuel handling system
- Ex-vessel fuel handling system
- In-vessel fuel handling system
- Nuclear island heating, ventilation, and air conditioning system
- Gaseous radwaste processing system

Source: Kemmerer Unit 1 PSAR (ML24088A065)

TerraPower Approach to PDC Development



[1] Use language from GDC, ARDC or MHTGR-DC, if applicable.

Topical Report Figure 1, “PDC Development Flowchart”

Natrium PDC – I. Overall Requirements

Criterion	Title	Basis DC	Modified?
1	Quality standards and records.	SFR-DC 1	Y – safety-significant
2	Design bases for protection against natural phenomena.	SFR-DC 2	Y – safety-significant
3	Fire protection.	SFR-DC 3	Y – safety-significant
4	Environmental and dynamic effects design bases.	SFR-DC 4	Y – safety-significant
5	Sharing of structures, systems, and components	SFR-DC 5	Y – safety-significant, safe shutdown

Natrium PDC – II. Multiple Barriers

Criterion	Title	Basis PDC	Modified?
10	Reactor design.	SFR-DC 10	Y – SARRDLs
11	Reactor inherent protection.	SFR-DC 11	N
12	Suppression of reactor power oscillations.	SFR-DC 12	Y – SARRDLs
13	Instrumentation and control.	MHTGR-DC 13	Y – coolant boundary
14	Primary coolant boundary.	SFR-DC 14	Y – coolant boundary
15	Primary coolant system design.	SFR-DC 15	Y – coolant boundary
16	Containment design.	MHTGR-DC 16	Y – safety-significant
17	Electric power systems.	SFR-DC 17	Y – safety-significant, SARRDLs
18	Inspection and testing of electric power systems.	SFR-DC 18	Y – safety-significant
19	Control room.	SFR-DC 19	Y – safe shutdown

Natrium PDC – III. Reactivity Control

Criterion	Title	Basis PDC	Modified?
20	Protection system functions	MHTGR-DC 20	Y – safety-significant
21	Protection system testability and reliability.	SFR-DC 21	N
22	Protection system independence.	SFR-DC 22	N
23	Protection system failure modes.	SFR-DC 23	N
24	Separation of protection and control systems.	SFR-DC 24	N
25	Protection system requirements for reactivity control malfunctions.	MHTGR-DC 25	N
26	Reactivity control systems.	SFR-DC 26	Y – SARRDLs
27	[None - incorporated into 26 consistent with RG 1.232]	N/A	N/A
28	Reactivity limits.	SFR-DC 28	Y – coolant boundary
29	Protection against anticipated operational occurrences.	SFR-DC 29	N

Natrium PDC – IV. Fluid Systems (1)

Criterion	Title	Basis PDC	Modified?
30	Quality of primary coolant boundary.	SFR-DC 30	Y – coolant boundary
31	Fracture prevention of primary coolant boundary.	SFR-DC 31	Y – coolant boundary
32	Inspection of primary coolant boundary	SFR-DC 32	Y – coolant boundary
33	Primary coolant inventory maintenance.	SFR-DC 33	Y – SARRDLs
34	Residual heat removal.	SFR-DC 34	Y – SARRDLs
35	Emergency core cooling.	SFR-DC 25	N
36	Inspection of emergency core cooling system.	SFR-DC 36	N
37	Testing of emergency core cooling system.	SFR-DC 37	Y – leaktight

Natrium PDC – IV. Fluid Systems (2)

Criterion	Title	Basis PDC	Modified?
38	[Not used – functional containment]	N/A	N/A
39	[Not used – functional containment]	N/A	N/A
40	[Not used – functional containment]	N/A	N/A
41	[Not used – functional containment]	N/A	N/A
42	[Not used – functional containment]	N/A	N/A
43	[Not used – functional containment]	N/A	N/A
44	Structural and equipment cooling.	SFR-DC 44	Y – safety-significant
45	Inspection of structural and equipment cooling systems.	SFR-DC 45	N
46	Testing of structural and equipment cooling systems.	SFR-DC 46	Y – leaktight

Natrium PDC – V. Reactor Containment

Criterion	Title	Basis PDC	Modified?
50	[Not used – functional containment]	N/A	N/A
51	[Not used – functional containment]	N/A	N/A
52	[Not used – functional containment]	N/A	N/A
53	[Not used – functional containment]	N/A	N/A
54	[Not used – functional containment]	N/A	N/A
55	[Not used – functional containment]	N/A	N/A
56	[Not used – functional containment]	N/A	N/A
57	[Not used – functional containment]	N/A	N/A

Natrium PDC – VI. Fuel and Reactivity Control

Criterion	Title	Basis PDC	Modified?
60	Control of releases of radioactive materials to the environment.	SFR-DC 60	N
61	Fuel storage and handling and radioactivity control.	SFR-DC 61	Y – safety-significant
62	Prevention of criticality in fuel storage and handling.	SFR-DC 62	N
63	Monitoring fuel and waste storage.	SFR-DC 63	N
64	Monitoring radioactivity releases.	SFR-DC 64	Y – functional containment

Natrium PDC – VII. Additional PDC

Criterion	Title	Basis PDC	Modified?
70	Intermediate coolant system.	SFR-DC 70	N
71	Primary coolant and cover gas purity control.	SFR-DC 71	N
72	Sodium heating systems.	SFR-DC 72	Y – safety-significant
73	Sodium leakage detection and reaction prevention and mitigation.	SFR-DC 73	Y – safety-significant
74	Sodium/water reaction prevention/mitigation.	SFR-DC 74	N
75	Quality of the intermediate coolant boundary.	SFR-DC 75	Y – safety-significant
76	Fracture prevention of the intermediate coolant boundary.	SFR-DC 76	Y – coolant boundary
77	Inspection of the intermediate coolant boundary.	SFR-DC 77	Y – safety-significant
78	Primary coolant system interfaces.	SFR-DC 78	Y – safety-significant, SARRDLs
79	Cover gas inventory maintenance.	SFR-DC 79	N
80	Reactor vessel and reactor system structural design basis.	MHTGR-DC 70	N
81	Reactor building design basis.	MHTGR-DC 71	Y – MHTGR-specific language
82	Provisions for periodic reactor building inspection.	MHTGR-DC 72	Y – MHTGR-specific language

A. Use of the term “safety-significant”

- Change: *Replace “important to safety” from RG 1.232 DC with “safety-significant” to align with language from NEI 18-04*
- RG 1.233: “Applicants referencing this RG are expected to use the terminology in NEI 18-04”
- DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap” (ML23277A139) identified that there may be some SSCs that may be “important to safety” but not “safety-significant” per NEI 18-04 process
  - No gap because of use of RG 1.232 DC (e.g., those related to managing and monitoring effluents resulting from normal operations)

B. Use of graded approach to coolant boundary quality

- Change: *Modified to indicate "safety-significant elements" of the primary or intermediate coolant boundary*
- Consistent with NEI 18-04 approach, not all elements of primary coolant boundary are considered safety-related (SR) *a priori*
  - Proper application of NEI 18-04 would appropriately classify structures, systems, and components (SSCs), resulting in quality, design, and performance requirements commensurate with safety significance
- SE notes that if primary coolant boundary components are not SR, an exemption may be needed from regulations

C. Use of SARRDLs

- Change: *SARRDLs used instead of specified acceptable fuel design limits (SAFDLs)*
- SARRDLs are compatible with Natrium design/licensing approach
  - High-reliability metallic fuel chemically compatible with coolant
  - Can establish fuel design limits as surrogates for SARRDLs
- SARRDLs are consistent with NEI 18-04 process that requires mechanistic source term evaluations
- SARRDLs provide appropriate performance-based approach to determining functional containment performance criteria
- Same basis as SARRDLs in RG 1.232
- **No staff determination on specific SARRDLs**

D. Use of functional containment concept

- Change: *Adoption of functional containment DC and non-inclusion of containment building DC*
- Per previous discussion on SARRDLs, functional containment is also compatible with Natrium design and NEI 18-04 process
  - Low-pressure operation
  - Margin to coolant boiling
  - Chemical compatibility between fuel and coolant
  - Lack of sodium-water interaction
- **No staff determination on specific functional containment barriers or performance**

E. Other generic changes

- Change: *Adoption of MHTGR-DC without MHTGR-specific language*
  - MHTGR language related to helium removed; no helium in Natrium
- Change: *Use of the term "safe shutdown"*
  - Sensitivity to "cold shutdown" for SFRs, coolant freezes at ambient temp
  - Change is consistent with RG 1.232, SECY-94-084, and NEI 18-04/RG 1.233
- Change: *Leak-tightness of cooling systems*
  - Anticipated in RG 1.232
  - Natural draft air circulation system used for emergency core cooling
  - Some amount of leakage not anticipated to impact ability of system to perform safety function