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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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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
(ACRS)

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GENERAL ATOMICS DESIGN SUBCOMMITTEE

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TUESDAY
MAY 2, 2023

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The Subcommittee met via Teleconference, at 12:30 p.m. EDT, Vicki M. Bier, Chair, presiding.

COMMITTEE MEMBERS:

VICKI M. BIER, Chair
RONALD G. BALLINGER, Member
CHARLES H. BROWN, JR., Member
VESNA B. DIMITRIJEVIC, Member
GREGORY H. HALNON, Member
WALTER L. KIRCHNER, Member
JOSE MARCH-LEUBA, Member
DAVID A. PETTI, Member
JOY L. REMPE, Member

ACRS CONSULTANTS:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

AGENDA

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

12:30 p.m.

CHAIR BIER: This meeting will now come to order. This is a meeting of the General Atomics Licensing Subcommittee of the Advisory Committee on Reactor Safeguards.

I am Vicki Bier, chairman of today's Subcommittee meeting. Here as members in attendance are David Petti, Charles Brown. Jose is here. Joy Rempe, Matt Sunseri, Ron Ballinger. Walt Kirchner I think will be back in a minute, probably. Greg Halnon is here.

Vesna, are you online? I can't really see.

MEMBER DIMITRIJEVIC: Yes, I am.

CHAIR BIER: Yes.

MEMBER DIMITRIJEVIC: Hi.

CHAIR BIER: Great. Thank you. And how about our consultants, Dennis Bley and Steve Schultz?

DR. BLEY: Dennis here.

CHAIR BIER: And it looks like Steve is also here. Apologies. I have to keep taking my glasses on and off for different distances. Okay.

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

During today's meeting the Subcommittee will review the staff's draft safety evaluation on the General Atomics Fast Modular Reactor principal design criteria. The Subcommittee will hear presentations by and hold discussions with the NRC staff, General Atomics' representatives, and other interested persons regarding this matter.

Parts of the presentations by the applicant and the NRC staff may be closed in order to discuss information that is proprietary to the licensees and its contractors pursuant to 5 USC 552 (b)(C)(iv).

Attendance in the meeting that deals with such information will be limited to the NRC staff and its consultants, General Atomics, and those individuals and organizations who have entered into an appropriate confidentiality agreement with them. Consequently, we will need to confirm that we have only eligible observers and participants in any closed part of today's meeting.

The rules for participation in all ACRS meetings, including today's, were announced in the Federal Register on June 13, 2019.

The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory

Committee Act.

For background, the ACRS is intended to be independent of the NRC staff. ACRS issues publicly available letter reports that provide the Commission our independent technical reviews of NRC staff evaluations of the safety of proposed reactor facilities.

It is required by the Atomic Energy Act that ACRS participate in the reviews of submittals for new reactor licenses. As part of our review, we consider not only the staff's safety evaluations but also the original submittals by the applicant.

As part of our review process, ACRS members will ask questions and at times make statements. However, these statements are individual member opinions and should not be construed as ACRS findings or opinions. ACRS opinions are only as documented in our written letter reports.

The ACRS section of the U.S. NRC public website provides our charters, bylaws, agendas, letter reports, and full transcripts of all full and subcommittee meetings, including the slides presented. The meeting notice and agenda for this meeting were also posted there.

So far we have received no written

statements or requests to make an oral statement from members of the public.

The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions as appropriate for deliberation by the full Committee.

A transcript of today's meeting is being kept and will be made available.

Today's meeting is being held in person and over Microsoft Teams for ACRS staff and members, NRC staff, and the applicant. There is also a telephone bridge line and a Microsoft Teams link allowing participation of the public.

When addressing the Subcommittee, participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. When not speaking, we request that participants mute your computer microphone or phone by pressing star 6.

We will now proceed with the meeting. And I'd like to start by calling up the NRR staff. And I believe that will be, sorry, Candace De Messieres. Sorry if I mispronounced that. Thank you.

MS. DE MESSIERES: Thank you, Chair Rempe and also Member Bier for the opportunity to present to the Committee today.

So I'm Candace De Messieres, Chief of the Advanced Reactor Technical Branch 2 in the Division of Advanced Reactors and Non-Power Production and Utilization Facilities, or DANU, in the Office of Nuclear Reactor Regulation.

Later in this meeting after the General Atomics design overview, the NRC staff will provide you with a summary of our review of the General Atomics Electromagnetic Systems, or GA-EMS, Fast Modular Reactor Principal Design Criteria Topical Report.

Like the light water-based general design criteria contained in Part 50, Appendix A, the PDC established the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components that are important to safety. Accordingly, generation of adequate PDC is a foundational step on the path to licensing.

In our review of the GA-EMS PDC Topical Report, the NRC staff leveraged the information in Regulatory Guide 1.232 that was reviewed by the ACRS in 2018 and provides guidance for developing generic advanced reactor design criteria, or ARDC, for technology specific sodium-cooled fast reactor and modular high temperature gas-cooled reactor PDC.

The staff also drew on its experience reviewing PDC for other advanced non-light water reactors, such as the Kairos Power fluoride salt-cooled high temperature reactor.

Thank you again for the opportunity to present to the Committee. And we look forward to hearing your insights and feedback later in the meeting. Thank you.

CHAIR BIER: Okay. I believe it is now time for the General Atomics introductory remarks by -- I'm not sure if that's -- oh, sorry, that's Aaron Majors I believe. And I don't know if you're in the room or online.

MR. MAJORS: I am online. Everyone hear me clearly?

CHAIR BIER: Yes.

MR. MAJORS: Thank you so much. I just want to start by saying thank you for taking the time out to have this review, very needed. And we're looking forward to hearing from the outcome of this meeting.

I'd like to say just a couple quotes that are apropos for safety. These authors are unknown.

But no safety, know pain, but if you know safety, you have no pain. Also, safety doesn't happen by accident.

And those are the safety moments that we live by here at General Atomics. General Atomics has a long history of developing extremely safe reactors. And this history began with the TRIGA research in reactors and has evolved into high temperature gas-cooled reactors is where we are today with our Fast Modular Reactor, which is an answer to a growing market and the need for small, easily deployable reactors that provide great stability through rapid load following.

And so our main objective is the achievement of proper operating conditions and the prevention or mitigation of accident consequences to protect our workers, the public, and the environment from radiation hazards.

So we're really happy to be here and looking forward to the outcome. Thank you.

CHAIR BIER: Okay. Thank you. We're happy to have you here.

So now the first part of the presentation is the overview of the General Atomics Fast Modular Reactor design. And I believe the presenter for that is going to be John Bolin. Is that correct?

MR. BOLIN: That's correct.

CHAIR BIER: Excellent. Welcome.

MR. BOLIN: Okay. Let's see. Is this displaying as just a single slide?

CHAIR BIER: Yes. We now see your slides.

MR. BOLIN: Okay. All right. So this is going to be an overview of the conceptual design to date. And I'll continue. And I am the safety and licensing lead here at GA-EMS for the Fast Modular Reactor.

So, before I go on to this goal, I wanted to introduce our team. We have a very distinguished team of collaborators, including a strategic partnership with Framatome, on this Fast Modular Reactor design. We have worked with Framatome in the past on the gas turbine-modular helium reactor and worked together and competed against each other on the next generation nuclear plant.

We also have on our team EPRI. And EPRI has, as part of their team, they have enlisted to help Vanderbilt University.

We also have two other universities that are collaborating with us, the University of Wisconsin-Madison, under the leadership of Mike

Corradini, and the University of Texas at Arlington. And they're focused on the turbine machine design.

We also have the expertise of three national labs, Idaho National Lab, with both their BISON and ATR and TREAT expertise. And we also have Argonne National Lab, with their fast reactor fuel design expertise, and Sandia National Lab, with their MELCOR modeling expertise.

So the goal is to develop a Fast Modular Reactor. It's 44 megawatts electric. And, you know, it's intended for flexible power generation and easily dispatchable and carbon free. And we're targeting commercial operations by 2035.

The team is developing key design attributes. It is a fast spectrum reactor. We use helium inert gas as coolant. We have pellet loaded fuel rods. We are emphasizing site flexibility and small passive heat removal systems that will result in safe, maintainable, and cost effective nuclear power generation.

The FMR project officially started on 15th of December of 2021. It's a three-year program under the ARDP, Advanced Reactor Concepts 2020 Program.

MEMBER REMPE: Hey, John. This is Joy. You know how ACRS members always are rude and interrupt people. I don't know if --

MR. BOLIN: And I was going to say you guys can ask me questions any time.

MEMBER REMPE: Things haven't changed over years. But anyway, I don't know if you're too close to the mic or there's some tapping sounds. But it's hard to sometimes hear what you're saying. Do you have an idea what it could be? And maybe --

MR. BOLIN: It might be my coffee flask is jiggling a little bit. So maybe that's --

MEMBER REMPE: Okay. That would help. Thank you. Sorry to interrupt. But it was getting distracting. Thanks.

MR. BOLIN: Okay. All right. We'll see if that's improved.

MEMBER REMPE: That is better. Thanks.

MR. BOLIN: Okay. I'll go on to the Next slide. So the project objectives, you know, their focus is to enable future deployment, development and deployment. And so we're particularly interested in verification of key metrics in fuel, safety, and operational performance.

So, as stated here, we will look at the technical feasibilities. I mean, basically the conceptual design effort is to prove the technical feasibility of the design, looking at high burn-up fuel operation, passive safety features, and rapid grid adaptability or load following.

The project obviously includes pre-application licensing activities with the NRC. That was a key desire of the DOE in their FOA. And the project will also conclude with an initial cost evaluation.

Like I said, the two focuses are on verification, both experimental and numerical verification. The experimental verification, we do have a fuel fabrication campaign that will result in a high burn-up irradiation test at, and transient test at ATR and TREAT to begin the qualification of the fuel design. And we'll go into that a little more in the later slides.

We also have scaled tests of the reactor vessel cooling system using the facility that University of Wisconsin-Madison has to further verify the passive cooling capability. In this case, the RVCS test facility that they have is actually between half scale and full scale of our RVCS design. So it will be a very interesting test.

We're also doing numerical verification. Part of this is the accident analysis work being done both, or being done at UW-M. They are developing a MELCOR model. So that will support the design work and pre-application licensing.

We are also doing, with Sandia, a MELCOR model to simulate the FMR plant and to demonstrate rapid load following capability, also load rejection and basically a variety of operational transients.

Any questions so far? Okay.

Okay. This is the, this goes over our effort to design the FMR core to improve safety margin. Some of the things to note on this slide is the core power density. Oh, and I should -- and so, in this slide, I'm comparing numbers for the Fast Modular Reactor, the gas turbine-modular helium reactor, also designed by General Atomics, and the AP1000 PWR.

So, I mean, the first state, of course, is the output. The reactor output is quite low. It's 100 megawatts thermal, you know, 6 times lower than the GT-MHR and much lower than the AP1000.

The power density is almost 15 megawatts per meter cubed. That's higher than the GT-MHR but much less than the AP1000.

The heat generated in the fuel, actually our number is similar to AP1000. Most of the heat does get deposited in the fuel, of course. Pressure is 7 megapascals.

The other thing to note, it's not explicitly mentioned here. So the outlet temperature is not, you have to calculate the outlet temperature based on these numbers. So the FMR has an outlet temperature of 800 degrees C, while the GT-MHR had an outlet temperature of 850 degrees C. So we cut back the outlet temperature a little bit to improve safety margin.

The other thing to note, of course, is the, similar to the power density, the fuel rod average linear power is quite low, much lower than the AP1000.

And the other thing is the fuel height.

DR. BLEY: John?

MR. BOLIN: Yes.

DR. BLEY: This is Dennis Bley.

MR. BOLIN: Yes, Dennis.

DR. BLEY: I'm remembering back a long time ago, probably from the '70s. Excuse me. You had a fast reactor design way back then. And if you lost force circulation, you had about 45 seconds I think, if my memory is right, to get it back to prevent significant damage. How does this reactor look if you lose force circulation?

MR. BOLIN: Well, there's two things that are in our favor. First off, we are using SiGA silicon carbide composite cladding. In fact, the whole fuel assembly is a ceramic composite cladding or ceramic composite material. And so it has a much higher temperature capability. But also we have greatly reduced the power density compared to the, I think you're referring to the gas-cooled fast breeder reactor --

DR. BLEY: That's probably true.

MR. BOLIN: -- back in the '70s. So the power density is much less.

And so, while I'm not going to present the accident results, the passive safety, we have engineered that so that we can safely cope with a loss of force circulation, loss of force cooling.

DR. BLEY: Okay. Thanks. I look forward to seeing more about that later.

MR. BOLIN: Sure. And like I said, so the cladding material is a SiGA silicon carbine composite. And we'll go into that a little bit more.

And the core height is also quite small compared to the other two designs shown here. It's only 1.8 meters in height. And this is the active height. This is the fuel zone of the fuel assembly.

Okay. Any questions before I move on?

MEMBER PETTI: John, this is Dave Petti. Are you going --

MR. BOLIN: Hi, Dave.

MEMBER PETTI: Hi. Are you going to show us some pictures of what a fuel assembly looks like?

MR. BOLIN: Yes, definitely.

MEMBER PETTI: Okay. Then I will wait. Thanks.

MR. BOLIN: Okay. In fact, it's, part of it is on the next slide here.

So the fuel design, it leverages both UO2 legacy fuel development and SiGA cladding development. So we purposefully chose high density UO2 that's been proven in LWRs and tested in fast reactors in order to minimize the fuel development timeframe.

The silicon carbine composite cladding, SiGA, it's undergoing testing and maturation to the DOE accident tolerant fuel program. And in fact, SiGA cladding is being irradiated presently in ATR.

The fuel design uses, actually uses the ATF-LWR dimensions, you know, so that the cladding is the same size as that being developed for ATF. But it does have, the fuel design does have a large plenum similar to what you find in the legacy liquid metal fast reactor fuel designs --

MEMBER REMPE: John. Oh, I'm sorry. Go ahead.

MR. BOLIN: Yes.

MEMBER REMPE: Okay. Did you finish your last sentence? I didn't mean to cut you off.

MR. BOLIN: Yes. Go ahead.

MEMBER REMPE: Okay. Well, I was curious if you could talk a little bit more about the end cap welding. There's an image shown here (audio interference) an end cap on with this SiGA material. And apparently you've made it through leak testing and pressure testing.

And how long has it been in the ATR? And how long is it scheduled to be in the ATR? Are they -- is it going through any PALM cycles in the ATR so it's sort of having some ramp testing?

MR. BOLIN: I don't know the details of that. I mean, we have made a lot of progress in the end cap welding. And these are sealed rodlets that have been hermetically tested and meet the hermeticity requirements. And they will go through a few cycles I believe. I don't know if it will go through a PALM or not. And I don't know the details of that.

MR. MAJORS: It's six cycles that our specimens will be in ATR.

MR. BOLIN: This is, I think she's particularly, Joy is particularly asking about -- that's -- and Aaron is correct. So the ATF, I don't know about the ATF cladding that's being irradiated. The FMR cladding will also be, go through, like Aaron said, it will go through up to six cycles.

MEMBER REMPE: But you've not started that test yet --

MR. BOLIN: That hasn't started yet.

MEMBER REMPE: Okay.

MR. BOLIN: I'm going to --

MR. MAJORS: It starts in December.

MEMBER REMPE: Oh, okay. So it starts this December. And what is the peak temperature that this end cap weld has survived to date?

MR. BOLIN: I don't know the answer to that.

MEMBER REMPE: Okay. I just am curious. I mean, it's not necessary for this PDR report, but, or PDC report, but I just am --

MR. BOLIN: Yes.

MEMBER REMPE: -- curious on how far, because I know that was an issue for a lot of years.

MEMBER PETTI: John, just a question on the diameter of the UO2, because you mentioned about fast and thermal. Is it the size of a thermal UO2 or a fast reactor UO2 or somewhere in between?

MR. BOLIN: Well, I believe the fast reactor UO2 was extremely small.

MEMBER PETTI: Right.

MR. BOLIN: So it's not like that. But --

MEMBER PETTI: Okay.

MR. BOLIN: -- density obviously is much less in the liquid metal fast reactor similarly. So the UO2 pellet diameter is a little bit smaller than a standard UO2 pellet, because we do have a somewhat larger gap between the pellet and the cladding --

MEMBER KIRCHNER: John, this is Walt Kirchner.

MR. BOLIN: It's basically the same as an LWR pellet.

MEMBER KIRCHNER: John, this is Walt Kirchner. So I'm thinking back to the prior work that GA did in this particular area. If I remember correctly, you were looking at uranium carbide pellets or platelets or different --

MR. BOLIN: Correct.

MEMBER KIRCHNER: -- designs, not UO2.

For this reactor with a longer lifetime, what is the, what's the effective full power years of the UO2 in terms of burn-up?

MR. BOLIN: I think I will cover that. But it's 100 megawatt days per --

MEMBER KIRCHNER: Metric ton?

MR. BOLIN: Yes.

MEMBER KIRCHNER: That's pretty high compared to the UO2 that's used, because --

MR. BOLIN: Currently licensed, correct. It's higher than what's currently licensed. Fast reactor oxide fuel tests get to that burn-up and higher. But --

MEMBER KIRCHNER: Doesn't it center quite a bit? I thought that's why you were looking at uranium carbide and not UO2 previously.

MR. BOLIN: Well, the reason we were looking at uranium carbide, and we still are pursuing that reactor design, the centering is going to be much lower than you might expect because the UO2, peak UO2 fuel temperature is much lower than LWRs. So, and I'll show you that. I think I show you that later. I might be getting my presentations mixed up.

But, you know, it's probably about, well, no, we'll see that, about 1,200 degrees C is the peak fuel temperature.

MEMBER KIRCHNER: And then the grid material for the X bundle is what?

MR. BOLIN: Silicon carbide composite.

MEMBER KIRCHNER: Silicon carbide as well.

MR. BOLIN: Yes.

MEMBER KIRCHNER: Thank you.

MR. BOLIN: And the support tube is also silicon carbide. And you can see a picture of silicon carbide composite cladding and in the X-ray tomography of a cladding tube, and then as Joy mentioned, the end cap welding, which we think we have perfected. So, and it's ready for testing.

MEMBER PETTI: John, the grid plate is also silicon carbide?

MR. BOLIN: Yes.

MEMBER PETTI: Thank you.

MR. BOLIN: Okay. This goes into more detail, a little bit more detail of the different steps we've gone through to prepare for the ATR and TREAT irradiation capsules. Like I said, we've enlisted Argonne's help in looking at the BISON fuel model, looking at fission gas release and swelling. And all those eventual fuel failure mechanisms are also part of their modeling efforts.

And so that's informed our fuel design and analysis and, in particular, the analysis of these test rodlets. We are looking at both standard size rodlets and reduced diameter rodlets.

And like I said, we're going to do both ATR irradiation for up to six cycles. And we've also designed the rodlets with different size gaps to look at performance, you know, both standard fuel rod performance and performance where there would be pellet clad interaction in possible failure. So that's a design into the analysis and the experimentation. So --

MEMBER PETTI: So, John --

MR. BOLIN: Yes.

MEMBER PETTI: -- just a question on the clad. If this gets proprietary, let me know. But, you know, the last time I looked at SiC-SiC cladding for ATF, there were some seminal papers out of Oak Ridge that, given the delta T across the clad, you get some pretty serious tensile stress built up because of differential or irradiation swelling across it. I would imagine the lower power density helps you with that --

MR. BOLIN: Correct.

MEMBER PETTI: -- delta T. So, but, you

know, they've gone to things like liners and stuff. This is just SiC-SiC, nothing special?

MR. BOLIN: It's the standard ATF silicon carbine composite cladding. So it has the monolithic outside layer and the composite woven silicon carbide fiber, infiltrated silicon carbide in the inner layer.

And you're correct. Certainly the, you know, our power density is much lower than light water reactors. So the thermal gradients are much lower. Also, operating at a higher temperature is actually, for silicon carbide is actually a benefit, too. So swelling --

MEMBER PETTI: Sure.

MR. BOLIN: -- swelling and irradiation damage is less at higher temperatures. So we have both of those factors in our favor.

MEMBER PETTI: Thanks.

MEMBER REMPE: Out of curiosity -- I'm sorry. Is someone else -- do you want to go first?

DR. SCHULTZ: That was me, Steve, Joy. You go ahead.

MEMBER REMPE: Oh, well, I was just --

DR. SCHULTZ: I'll come in next.

MEMBER REMPE: Okay. I was curious about the instrumentation and what you're trying to validate with these, or verify with these tests. Are you, is it just temperature or are you -- and then post-irradiation examinations or are you going to try and do any other type of measurements online that --

MR. BOLIN: No, no other --

MEMBER REMPE: -- tests?

MR. BOLIN: No other measurements online. But we will look at, post-irradiation examination we'll look at fuel physical changes and fission gas release. So that will be looked at.

Actually, it's in that box right there. The PIE will look at fission gas release and fuel and cladding deformation. And particularly in the cases where we have reduced gap between the fuel and the cladding, you know, there's a possibility of cladding fracture that also needs to be looked at.

MEMBER REMPE: Will you have temperature instrumentation in the tests themselves?

MR. BOLIN: The fuel itself will not be temperature monitored, no.

MEMBER REMPE: Okay. And just to caution --

MR. BOLIN: At least not in the ATR capsule. I don't believe it's in the ATR capsule. The TREAT capsule may have instrumentation for that transient test.

MEMBER REMPE: In ATR sometimes small geometry changes that are within specifications can lead to interesting changes in temperatures that you don't expect. It's just a caution.

Anyway, Steve, go ahead.

DR. SCHULTZ: My question was related, John. And that is, in the testing, are you going to achieve those temperatures that you anticipate in the, for the reactor design parameters? The first question.

MR. BOLIN: Yes. In fact, we will have higher fuel temperatures than FMR will experience. We'll have higher temperatures.

DR. SCHULTZ: Good. And for the six cycles of operation, what burn-up do you expect to achieve in the fuel test?

MR. BOLIN: We will get close to 100 megawatt days for burn-up.

DR. SCHULTZ: Good. Thank you.

MR. BOLIN: It's a very, unfortunately I think, but it is a very accelerated test.

DR. SCHULTZ: It certainly appears that way. Thank you.

MR. BOLIN: Yes.

This just goes over the FMR test rodlets. And like I said, they are being fabricated. Actually, the rodlets have been fabricated. And they are going to be loaded with UO2 pellets actually next week. And then they'll be, the final end cap will be welded on and shipped to INL for insertion into ATR at the end of the year. So the fuel pellet processing is basically standard UO2 fuel pellet processing.

The other steps are part of the silicon carbide composite cladding fabrication. The silicon carbide fiber is braided together, then infiltrated with silicon carbide, and then both infiltration and then deposition of an outside silicon carbide layer. Pellets are then loaded. And then the final end cap is sealed. So this is obviously a key accomplishment of our conceptual design effort is to actually make these fuel rods and to have them tested.

We were particularly, it was particularly important to us to not just do a paper study on conceptual design, but to actually do, like we mentioned earlier, experimental verification of the design.

The other, of course, part of our defense in depth is the vessel system. We have a, the vessel is sized, you know, for normal operation AOO conditions. The design code, the ASME design code, of course, is used. That is Section 3, Division 5, the 2021 edition.

Right now the thickness is adequate for 300,000 hours or 35 effective full power years based upon the code. The code data does suggest very little change out to, because of the temperatures that we're at, very little change out to 60 years. And so a future code revision should not have an impact on our vessel design.

But we're also, one of the key problems with gas-cooled reactors, helium gas-cooled reactors is, of course, helium leakage. And so we pay particular attention to using seal welds at all joints to minimize helium leakage.

And another interesting thing is that a lot of accidents and even load following, you know, involve flow reductions. As we'll go over on the next slide, we'll see, I'll discuss about the flow reductions through normal operation. But all these events, because we're using a Brayton cycle, the pressure load on the vessel decreases during these flow reductions.

MEMBER MARCH-LEUBA: John, this is Jose March-Leuba. Just a layman question, you mentioned earlier an exit gas temperature of 800 degrees C.

MR. BOLIN: Correct.

MEMBER MARCH-LEUBA: What materials are you using for the hot leg and the vessel? And what temperatures do they have to survive?

MR. BOLIN: Well, so, like the GT-MHR, we do have a cross vessel that connects the reactor vessel to the power conversion unit. And so it has, the hot gas in on the inside of this cross vessel. There's an insulated layer on the inside of this cross vessel that then protects.

And then we have cold helium. Cold is a relative term, you know. It's 509 degrees C on the outside of this duct. And, you know, the layer that connects to the cross vessel sees that 509 degrees C. So all of the vessel materials are 509 degrees C or lower.

MEMBER MARCH-LEUBA: The gas is the one that has 7 megapascals. Somebody has to contain the helium. I hope you have thought through this. I don't know anything about this, but --

MR. BOLIN: It certainly is something we have dealt with on numerous, particularly the GT-MHR gas-cooled reactor design has looked at this extensively.

MEMBER MARCH-LEUBA: Yeah, and another question I know even more about. You don't mention anything about reactivity control. How do you plan to control reactivity?

MR. BOLIN: We have both control rods, I think boron carbide control rods and shutdown rods. They will have also silicon carbide cladding.

MEMBER MARCH-LEUBA: But they're not shown here in the picture, right? I don't see --

MR. BOLIN: No, no. Just the upper drive mechanisms are shown there.

MEMBER MARCH-LEUBA: Yeah, one important concern when you go for the final certification will be, priming along is the control rod has to be a design, has to have a design temperature that is higher than the fuel. In other words, you should not have an accident when you can meld a control rod and leave the fuel intact, because that would be bad.

MR. BOLIN: Yes.

MEMBER MARCH-LEUBA: So you're saying your design --

MR. BOLIN: That's why we are using silicon carbide cladding for the control rods. I don't know. Yeah.

MEMBER MARCH-LEUBA: And you said boron

carbide inside?

MR. BOLIN: Yes.

MEMBER MARCH-LEUBA: Okay.

MR. BOLIN: I guess my final comment was that the conceptual design has been completed on the reactor vessel internals. We're still working on the details of the power conversion system, which I'm going to discuss on the Next slide. And also this shows the arrangement of neutron shields around the core, of course. And there is a core shroud that protects the vessel top head from the high temperature gas exiting the core. So that's also an insulated layer that protects the top head.

Cold helium coming into the reactor goes all the way around the vessel and down the outside core barrel and into the lower portion of the vessel head and then up through the core.

MEMBER PETTI: John?

MR. BOLIN: Yes?

MEMBER PETTI: Just a question on your outer reflector. Is it stainless steel like in sodium systems? Or I know you guys had a design once with beryllium carbide as an outer reflector.

MR. BOLIN: No, we're using -- zirconium silicide is our reflector that's adjacent to the fuel.

It is zirconium silicide. It's a product that we're developing that we think it is a better reflector. I mean, it's not as good as stainless steel, but, fortunately, it has higher temperature capability. And it helps to minimize, then, fuel rod peakings along the reflector edge.

MEMBER KIRCHNER: And the upper and lower reflectors, are they the same or --

MR. BOLIN: The reflector that is right next to the fuel is always going to be zirconium silicide. Now, the outside reflector, outside of the zirconium silicide, we'll be using graphite, and I think, also, on the bottom.

MEMBER KIRCHNER: Upper and lower reflectors are graphite?

MR. BOLIN: Well, below -- like I said, there's always a zirconium silicide layer immediately next to the fuel, the core. So, both the upper and lower part is, first, zirconium silicide, and then, graphite. In the core, I don't think that's the case, but in the outer reflector that's the case. And the lower reflector, that definitely is the case, yes.

MEMBER PETTI: But the inner reflector has, like, an annulus of graphite?

MR. BOLIN: Yes -- no, no. The inner

reflector is always zirconium silicide.

MEMBER BALLINGER: This is Ron Ballinger.

I understand 300,000 hours, but I don't understand 540,000 hours. You say, "Code revision." Is this talking about Division 5 again?

MR. BOLIN: Yes, but, right now, the material, 316 stainless steel, is only allowed up to 300,000 hours.

MEMBER BALLINGER: Right.

MR. BOLIN: And so, future ASME Code revision is intended to extend that to 540,000 hours.

MEMBER BALLINGER: And that's in process?

MR. BOLIN: Yes. And the data, of course, already exist and shows very little change between 300,000 hours and 540,000 hours. So, it's not expected to have any design impact.

MEMBER PETTI: What's the vessel material again?

MR. BOLIN: 316 stainless steel.

MEMBER PETTI: Okay.

MR. BOLIN: Okay. So, the next slide goes over a little bit on the power conversion system. This is fairly standard for gas turbine design. It is a direct Brayton cycle. I know it's maybe hard to see. And I don't go into the details on the core side.

So, exit temperature from the reactor goes to the turbine, where, obviously, it goes down in temperature and pressure. It goes through the recuperator. So, the turbine outlet temperature is, basically, directing the reactor inlet temperature. So, there's a heat exchange between these two fluid streams.

From the recuperator, it goes to a pre-cooler that cools the helium before going to the low pressure compressor. And we have an intercooler in this design. So, from the low pressure compressor, you go to the intercooler. So, the heat that is added during the compression process in the low pressure compressor is removed by the intercooler, and that, then, goes to the high pressure compressor, and then, goes to the recuperator, and then, to the reactor.

So, this provides a high efficiency. And like we said, it's the net -- the electrical output from the generator is 44 megawatts electric.

The other interesting thing that enables rapid load following is that we are using a GA product, a current magnet motor generator. It is a variable frequency generator. So, it can change its rotational speed as needed. So, we can change the flow rate to the reactor by controlling the frequency and speed of the current magnet motor generator.

It goes through an AC/DC-AC frequency converter to get to the grid frequency. And, of course, that goes to the grid.

And so, this combination allows us to change the speed of the generator, change the speed of the turbine machinery, and thereby change the flow rate through the core. But, at the same time, while we're doing that, we're maintaining the frequency that's being fed to the grid.

And so, it promotes both rapid load following -- we're shooting for 20 percent per minute load following changes, but, also, it promotes grid stability. So, those are two things that we see as being important to the market in the future, and particularly, as the electric market gets more and more intermittent renewable energy sources.

MEMBER MARCH-LEUBA: And this is Jose again.

With those power rates 20 percent per minute, have you analyzed what happens to the UO2 pellets?

MR. BOLIN: So, that is going to be part of our modeling that is still ongoing.

MEMBER MARCH-LEUBA: Yes, because in the life of the reactors, if you ramp up like that, you will blow up the zirconium.

MR. BOLIN: Well, you know, the --

MEMBER MARCH-LEUBA: Yeah.

MR. BOLIN: Yeah. And, I mean, there's two things that are in our favor: lower power density and the silicon carbide is quite strong. And although sometimes it's viewed as a negative, the fact that it's not very ductile also means it keeps its shape, even if the pressure inside is increasing.

MEMBER REMPE: John, I just assumed that the TREAT test would encompass that. That's not going to -- you're not going to try to run the presentient (phonetic) test to such changes?

MR. BOLIN: The TREAT test is a reactivity-initiated accident. So, it is doing that, a reactivity-initiated accident simulation, not a load following test.

MEMBER REMPE: But the ramp and the change in power, wouldn't that --

MR. BOLIN: It should bound any reactivity -- it should bound any load following change.

MEMBER REMPE: Yes, I would think it would. Either that or -- it seems like you would want to test that before you start doing load following in the reactor.

MEMBER BALLINGER: This is a very different paradigm. I mean, we've got to stop thinking about zirconium cladding, because the zirconium, that's what the load following problem was for in light water reactors. It was for zirconium cladding. It only takes 10 or 20 degrees C to crack the UO2 at delta T. So, it's really the silicon carbide that's got to take it, and it's very rigid compared to zirconium, and there's no environment.

So, it's a different way of having to think of it. It requires a heck of a lot more data and experiments, but it's a very different fuel system.

MEMBER REMPE: Yes, but I just think I would rather test it out of the reactor.

MEMBER BALLINGER: I'll agree with you.

(Laughter.)

MEMBER REMPE: Yes, ahead of time.

MEMBER BALLINGER: Yes.

MEMBER REMPE: And I would find some way of doing it.

MEMBER BALLINGER: Yes.

MEMBER REMPE: But it's not included in

your plans?

Also, there's a lot of feedback, and I don't know -- John, again, there's something with your system, it looks like, according to the computers here in the room.

MR. BOLIN: Oh, well, I mean, it could be the fan in my laptop. So, I'm not sure I could control that.

MEMBER REMPE: We did have that happen with one of our members. Don't turn off your computer. But, anyway, it started up recently, yes, but anyway, you're going to have to have your managers buy you a better computer.

(Laughter.)

MEMBER REMPE: But, anyway, yes, so you do not have any plans to try and -- you're just going to do it by analysis?

MR. BOLIN: Just do it by analysis for now, yes.

CHAIR BIER: While we're paused here, I also to check and see if Aaron had any comments or clarifications that he wanted to add. Because anything in the chat does not make it into the transcript and the public meeting.

I guess maybe not right now.

MEMBER KIRCHNER: So, John, this is Walt Kirchner.

So, on this one, you're probably not going to try and put a bottoming Rankine cycle on this?

MR. BOLIN: No.

MEMBER KIRCHNER: No? Okay.

MR. BOLIN: The temperature coming out the recuperator is on the order of between 150 and 200 degrees C. So, there's not much left to take out.

MEMBER KIRCHNER: Okay. Right.

MR. MAJORS: Chairman? Chairman, my apologies, I was on mute talking. I'm sorry.

CHAIR BIER: Okay, go ahead, Aaron.

MR. MAJORS: I was just following up. Someone had asked the question about the peak temperature survival for the end caps. And I just wanted to add -- I didn't get a chance to interject because you guys were rapid-firing questions, which is great. We're trying to transcribe all these questions as well for our technical team. But I just wanted to add some points of reference.

So, our high-temperature gas application for joining our end caps is sufficient for normal operating and accident performance. So, our end caps are stable up to 1900 degrees Celsius for inert environments and 1750 for steam.

MEMBER REMPE: Thank you.

CHAIR BIER: Any other questions or comments for Aaron?

MEMBER KIRCHNER: Could you share who your fuel vendor is, or is that TBD?

MR. MAJORS: Fuel vendor is TBD.

MEMBER KIRCHNER: TBD? So, it's not one of the fly rod manufacturers? I'm thinking of Orvis, Sage, Winston. This is a joke. The technology that you're describing is what's used to make graphite fly rods for the cladding.

MEMBER BROWN: They'll use Loomis.

MEMBER KIRCHNER: Loomis? Okay.

(Laughter.)

MEMBER REMPE: And so, maybe this was in the reading material that I've forgotten, but this variable frequency generator, has anyone used it? Has it been built? What's its status?

MR. BOLIN: This is a proven product at GA-EMS. We have built an 8-megawatt version of both the current magnet motor generator -- and the frequency converter, we build that in modular fashion. So, we have, I believe -- and, Aaron, you can correct me -- I think we have 1-megawatt electric modules that we have built. And so, they're assembled in a modular fashion.

And so, for the 44-megawatt, we will be scaling both the current magnet motor generator up and the frequency converter up. But we have already proven the design at a reduced scale.

MEMBER REMPE: Okay. Thank you.

MR. BOLIN: And also on this slide, I wanted to address -- we purposefully chose to use dry cooling, even though that does have an efficiency penalty. But that, clearly, will reduce the impact on water resources and expands our siting options, particularly, if you consider that a lot of the solar and wind generation is going to be in possibly dry areas of the West. So, that was a deliberate selection on our part.

MR. FAIBISH: John, can I chime in? This is Ron Faibish with General Atomics. Can I chime in on something about silicon carbide?

MR. BOLIN: Sure.

MR. FAIBISH: I just wanted to add -- there were a lot of questions about the cladding. And we have had campaigns at HFIR in Oak Ridge on irradiation and prototypical conditions of 800 degrees C of (audio interference) and outlets.

So, variability was both shown -- and this was, actually, a campaign back for EM-Squared back when we were actively pursuing that design. The outlet temperature of 800 degrees C is very applicable, obviously, for FMR.

And also, in addition to that, as John mentioned, there is a campaign starting at ATL to do additional testings. So, that's up and coming.

But I just wanted you to know that silicon carbide has been exposed to irradiation and to high-temperature conditions and showing good results. And I think we're going to get more information to you, as needed, from previous tests. So, I just wanted to chime in on that.

Thanks, John.

MR. BOLIN: Thank you, Ron. Okay. Let's move on. Well, let's see.

And, obviously, in a defense-in-depth, you know, the third barrier we have is the containment. Now, this is, unlike a lot of gas-cooled reactors, this is, actually, a leak-tight containment. It is below grade, like most gas-cooled reactors have been below grade, but it is a leak-tight containment.

This was a deliberate selection to prove safety and siting. Obviously, it also has an impact, then, on fuel development and qualification.

Clearly, with TRISO fuel, there's a high standard that TRISO fuel has to meet. And by having a containment, we provide that extra defense for potential fuel failures during severe accidents.

This also shows the arrangement of the reactor. I don't know how I can get a pointer on this. Let's see. There we go. Can you see that? Yes, you can.

So, obviously -- maybe not obvious -- this was the arrangement that was presented in the proposal. So, it doesn't reflect concept design work to date, but here is the reactor vessel. And around the reactor vessel is the reactor vessel cooling system, which we'll cover in the Next slide. And then, in another compartment is the power conversion unit, the power conversion vessel. We also have -- which will also be covered in the next slide -- the maintenance cooling system. So, that is an active forced cooling system that is provided in case the power conversion system is unavailable. It is non-safety-related. Well, I'm getting ahead of myself.

And also, above --

MEMBER PETTI: John, functionally, it's like the shutdown cooling system in a HTGR?

MR. BOLIN: Correct. Correct.

MEMBER PETTI: Thanks.

MR. BOLIN: Functionally, it's like the shutdown cooling system.

It is also like the direct reactor auxiliary cooling system of liquid metal reactors, and EM-Squared had that kind of a system, a forced cooling system.

And up above here, we have RVCS water tanks, which I'll also discuss in the Next slide. So, this is just a general arrangement of the structure, the below-grade structure, of the containment. And it's a Category 1 structure.

The need for containment heat removal, cleanup, and venting, those are still under investigation. It is something addressed as a possibility in the PDCs as something that might be necessary, but it's still under investigation.

Obviously, below-grade containment is intended to also make us less vulnerable to airplane crashes.

MEMBER PETTI: John, I wanted to go back for a minute to the Brayton cycle.

MR. BOLIN: Go back to that slide?

MEMBER PETTI: No, no, just a question on it.

On the bigger machines, there was always a lot of development to get it to work, but at these smaller sizes, I know that you can get such components for gas systems. What's the status for helium systems? Are they commercially available or?

MR. BOLIN: No, they're not commercially available. We have not identified a manufacturer. Clearly, there are a variety of turbine manufacturers that we could choose from, but no one is building helium turbine machinery. Obviously, there are air-driven --

MEMBER PETTI: Right.

MR. BOLIN: -- turbine machines, but no helium ones.

MEMBER PETTI: Thanks.

MR. BOLIN: Sure. Okay. So, like I alluded to, we have residual heat is being removed by both active and passive systems. Really, the first line of defense is the power conversion system itself. We are, in particular, designing the system so that, if there is a grid disruption, that the power conversion system will ramp down and provide house loads and cool the reactor at house loads.

But if the power diversion system is the source of the problem, then the maintenance cooling system is available to remove core residual heat after reactor shutdown. And so, like we said, it is similar in function to the shutdown cooling system. Basically, it's taking hot helium out of the reactor, bringing it over helium-to-water heat exchanger, and then, circulating that back into the reactor, and basically, cooling the core. Details are still being worked on.

And so, that water is cooled in a cooling tower by forced air. So, all that system is intended to be not safety-related.

The safety-related system is the RVCS, although it has, actually, also has a non-safety-related component to it. So, the RVCS, the Reactor Vessel Cooling System, has two loops. The panel of two loops surrounding the reactor has alternating tubes of one loop or the other loop.

The water in the RVCS circulates naturally by buoyancy-driven flow. The water goes into a tank. So, this tank -- there's two tanks. Like I say, everything is redundant. There's also two of these cooling towers, and both of these cooling towers have a -- there's a heat exchanger in the RVCS tank that is cooled by this cooling tower. So, it keeps the water in the tank cold during normal operation -- both tanks. Like I said, there's another cooling tower with heat exchangers that's cooling this tank.

In an accident where we lose power, you know, for the safety-related portion of the system, the cooling tower is not safety-related. The pump for this water is not safety-related. So, this whole cooling system of the tank is not safety-related. So, during an accident where we lose all non-safety-related systems, then the RVCS loses heat by boiling off water from the water tank. And the water tank is sized so that we have seven days of boil-off with just one loop. And obviously, if we have both loops that are functioning, then we have, you know, much longer capacity to cool the reactor and vessel system.

Also, like many gas-cooled reactors, we have an annular core arrangement that promotes passive heat removal from the core through the reflector. This, actually, like I said, this shows the zirconium silicide reflector -- is this green. The blue is also zirconium silicide. The central zone is also zirconium silicide. And there is a graphite outer layer outside of the zirconium silicide.

And there's three, yes, three fuel zones. We have a three-batch core. And every 15 or more years, we have a refueling and we replace one-third of the core.

Let's see. I think that's all I need to say about this slide. Any questions on this slide?

MEMBER KIRCHNER: So, John, this is Walt Kirchner again.

MR. BOLIN: Yes?

MEMBER KIRCHNER: So, on this MCS, then, you said that's not a safety-grade system. So, you must have isolation valves to and from the helium circuit?

MR. BOLIN: It is a --

MEMBER KIRCHNER: Because you have a water heat exchanger there.

MR. BOLIN: We have isolation valves on the waterlines, definitely. But the MCS is in the containment. So, we have a --

MEMBER KIRCHNER: So, you're designing for the potential that you would have a break in that line --

MR. BOLIN: We'd have a break in --

MEMBER KIRCHNER: -- and the containment would have the pressure rating --

MR. BOLIN: Correct.

MEMBER KIRCHNER: -- to withstand that

blowdown? Okay.

MR. BOLIN: Correct. So, the MCS is one of our sources of primary coolant breaks. Now, there is a flow shutoff valve downstream -- it will probably be downstream of the circulator. Because, otherwise, we'll get natural circulation through this maintenance cooling system. So, we have a flow shutoff valve for normal operation.

MEMBER KIRCHNER: Thank you.

MR. BOLIN: Okay. This is my last slide. Or, no, it's not my last slide. Okay.

So, one of the things that has been a concern with gas-cooled reactors is bypass flow. And also, related to that is flow-induced oscillations. I know that there is a PDC on power oscillations.

The coolant itself, of course, it doesn't have a reactivity effect, but movement of core and reflector structures can have some reactivity effect. In particular, the bypass flow, you know, because the -- I don't know if it was clear, but the fuel assembly is relatively open. So, it's kind of an open bundle. It does have brackets on the corners, but, basically, as the flow comes up through the fuel assembly, it can redistribute among the various fuel assemblies.

But, also, because it's open, it can also feed bypass flow paths. So, as you see here, around the outer fuel assemblies, and also, all throughout the inner fuel assemblies, there are gaps between these blocks. And so, because of the open fuel assembly, we can get bypass flow going from the fuel area into the central zone and into the other reflector zone. And, in fact, this outer green zone also has gaps in it, because it's going to be in pieces.

And so, all these gaps contribute to bypass flow. It has a benefit, though. These bypass flow paths also improve heat transfer during loss-of-forced-cooling accidents. So, while our predominant heat transfer method is by radiation heat transfer from the fuel assemblies to the reflector, we also get a natural circulation from the upper plenum down through the outer reflector bypass flow paths, and then, into the fuel and back up, into the fuel here and back up.

So, the bypass flow has a negative effect during normal operation because it reduces the amount of flow through the core, but it has a positive effect during loss-of-forced-cooling accidents because it aids in the natural circulation of helium through the core.

Any questions on this?

MEMBER KIRCHNER: Can you give us, John -- this is Walt Kirchner -- just some estimate of steady state? What's the centerline temperature in your peak rod bundle, roughly? You know, compared to LWR, the cooling is not as efficient as a forced-water circulation system. So, I'm imagining that your delta T that's building up with the UO2 pellet on the order of diameter of an LWR fuel rod would have a centerline temperature that's running significantly higher. Is your power density so low --

MR. BOLIN: Let's go back --

MEMBER KIRCHNER: -- versus your surface area, that the centerline uranium oxide temperatures are low?

MR. BOLIN: Well, let's go back to the chart I had.

So, yes, you can see here that the fuel rod linear power is about -- it's not quite 10 times lower, but it's close, almost 10 times lower linear power than an LWR. So, the delta T in the fuel is, correspondingly, 10 times lower. And so, our peak fuel temperatures are running around, I think around 1200 degrees C.

MEMBER KIRCHNER: Okay. It doesn't quite

scale like that. I'm sorry.

MR. BOLIN: Well, it doesn't quite --

MEMBER KIRCHNER: You've got a helium coolant and --

MR. BOLIN: The helium -- the cladding temperature is a lot higher.

MEMBER KIRCHNER: Yes.

MR. BOLIN: Yes, the cladding temperature is -- obviously, the cladding temperature is higher than the helium that's cooling it. But, if the cladding temperature is around 800 degrees C, we have, like I said, a much lower fuel delta T, and it, correspondingly, lowers our peak fuel temperature.

MEMBER KIRCHNER: It would be useful to have simple graphs at least for steady-state full-power conditions -- what your centerline UO2 temperature; what your cladding temperature is; what the coolant is.

MR. BOLIN: Yes, I have that.

MEMBER KIRCHNER: And then, for a loss of -- I think Dennis asked earlier -- for a loss of forced circulation, where this was a real issue for some of the fast reactors, gas-cooled fast reactor designs, what kind of temperature excursion you would see in a loss-of-flow condition?

MR. BOLIN: Yes, I think that is a subject that is best addressed later, I mean in a subsequent discussion maybe. I actually do have some slides on that, but it is for a different meeting. So, I didn't put it in this package.

So, for accident conditions, we are --

MEMBER KIRCHNER: These comparisons that you're showing aren't very useful, actually.

MR. BOLIN: Well, yes, I mean, they're --

MEMBER KIRCHNER: Because I think you get my point. I mean, it's a rather complicated thermal hydraulic set of conditions that doesn't extrapolate well from gas to water.

MR. BOLIN: Correct.

MEMBER KIRCHNER: It depends on how much surface area you have; what your ilium flow velocity is --

MR. BOLIN: So, the surface --

MEMBER KIRCHNER: -- a whole number of things. So, when you just do this apples-and-oranges comparison, it's not terribly useful. It would be much more for your benefit in making your case to show what the actual operating conditions are for the fuel and the cladding, and how the system responds in a loss-of-flow condition.

MEMBER MARCH-LEUBA: Yes, this is Jose.

Eventually, you're going to have to choose your licensing basis events and run all the Chapter 15 analyses and figure out what your temperatures are everywhere. But, before I invest money in the design, I assume you have run some preliminary calculations for what you consider to be the limiting event.

MR. BOLIN: Correct, we have.

MEMBER MARCH-LEUBA: And I would expect that likely to be loss of pressure and when you lose your gas.

MR. BOLIN: Correct.

MEMBER MARCH-LEUBA: And then, you started cooling off by radiation. What temperatures you reach -- are you going to be okay?

MR. BOLIN: Correct, we are. We've designed it for that accident.

And as far as the surface area is concerned, remember, our fuel rods have the same geometry as the light water reactor fuel rods. So, our fuel rod surface area is the same as the light water reactor.

But that's about, you know, in the future, you will need to present both normal operation and accident condition fuel performance.

So, we have -- and I know the NRC project management is also going to go over this -- but we have made a lot of progress in pre-application licensing with the NRC. We have prepared a Regulatory Engagement Plan, you know, that outlines our licensing strategy for this conceptual period.

Obviously, the subject of this discussion is the principal design criteria. We submitted that; got some requests for additional information, and we responded to those and revised the Topical Report. And that's the subject of this meeting, is the NRC's Safety Evaluation on that Topical Report.

We have also submitted a QA program description Topical Report. We got some feedback early on, and then, made changes to that document. And that's undergoing review now.

And we've also submitted a Fuel Qualification Plan Technical Report. It's still a fairly early document. It's like a white paper. It's not a Topical Report.

And then, we have other documents that are planned: a Source Term Methodology, LBE selection, PRA, and safety classification.

CHAIR BIER: A quick question here. You're expecting that the safety classification decisions will be made based on the PRA or they are two separate processes?

MR. BOLIN: Well, we will not have a complete PRA at the end of this conceptual design period. But the safety classification will be informed by the preliminary risk assessment work that is being done by, actually, being done by Vanderbilt University -- and also, historical PRA work that we've done on gas-cooled reactors in the past. So, it will be risk-informed safety classification. And the LBE selection will also be risk-informed, but not based on a complete PRA.

CHAIR BIER: Okay. Thanks.

MEMBER KIRCHNER: John, this is Walt Kirchner.

MR. BOLIN: Yes?

MEMBER KIRCHNER: I started to digress or regress. Your basic reactivity control in terms of accident conditions is based on leakage? In other words, the diameter of your core and power level were chosen or are pretty much indirectly determined depending on leakage?

You've gone away from reflector control to control rod controls. So, in an offsite condition, what's the primary shutdown mechanism for this? Is it

Doppler and leakage? Could you give us just a feeling for your core design philosophy in terms of reactivity control?

MR. BOLIN: Well, while Doppler is a factor that can lessen the reactivity control requirements, we are still primarily relying on control rods and shutdown rods. So, we have control rods that will be, you know, partially inserted into the core for power adjustment, but we also have shutdown rods that will be fully removed from the core that will be used for shutdown, out-of-steam shutdown rods.

So, leakage and Doppler and reactivity coefficients, those will all play a factor, but we're still primarily relying on control rods and shutdown rods.

MEMBER KIRCHNER: I'm thinking to offsite conditions.

MR. BOLIN: Yes.

MEMBER KIRCHNER: So, our fast sodium reactors, typically, are relying on leakage -- that determines the maximum size of the core -- and expansion.

MR. BOLIN: Yes, we're not relying on expansion or --

MEMBER KIRCHNER: Right.

MR. BOLIN: Like I said, leakage is not -- I mean, it's there, but we're not relying on leakage.

MEMBER KIRCHNER: Are your reflectors, your radial reflectors, positive or neutral? Or negative?

MR. BOLIN: I don't know the answer to that.

MEMBER KIRCHNER: It's just something to think about.

MR. BOLIN: Okay.

MEMBER KIRCHNER: We'll ask in a future engagement.

MR. BOLIN: Yes.

MEMBER KIRCHNER: Thank you.

MEMBER PETTI: So, John, then, what determines the size? Why is it 110 megawatts? Was there some accident that limited, you know --

MR. BOLIN: Definitely. It definitely was the depressurized loss-of-forced-cooling accident that determined the size. And, in fact, we originally were looking at 50 megawatts electric and 120 megawatts thermal. And we ended up reducing the power to 100 megawatts thermal for the depressurized loss-of-forced-cooling transient.

MEMBER PETTI: And is it a vessel temperature or a peak silicon carbide temperature?

MR. BOLIN: It was peak silicon carbide temperature.

MEMBER MARCH-LEUBA: Am I to assume from what you said that your safety margin is 20 percent? I mean, you cannot handle 50 megawatts electric, but you can handle 40? I mean, that's very limited for an advanced reactor.

Yes, I'm just putting it in the record. You don't have to answer it. But we're used to seeing reactors that you can shoot them with a shotgun and nothing happens to it.

MR. BOLIN: Yes.

MEMBER MARCH-LEUBA: Are you telling me that you want to go out to 20 percent power operate and make it?

All right. Don't answer. I'll put the bad things on the record, but you don't have to --

MR. BOLIN: I would say that the analysis has improved during the conceptual design period, but we have decided to stick with the 100 megawatts rather than try to minimize our margin.

And silicon carbide has different --

MEMBER KIRCHNER: Just to calibrate us,

John, what do you use as a benchmark for your silicon carbide structure as a thermal, like a thermal limit, to determine, you know, it remains intact in a loss-of-depressurization and loss-of-forced-circulation event?

Because that's, typically, you know, if you go back to the HTGR business that you all were involved in, you know, sizing of the core was kind of an inverse calculation of temperature of the vessel and temperature of the TRISO particle fuel.

What limits here? If it's silicon carbide, can you calibrate us? What temperature is that?

MR. BOLIN: The temperature limit we have been using is 1800 degrees C. So, we want our silicon carbide to be below 1800 degrees C.

MEMBER MARCH-LEUBA: And that's because of the welding on the top of the rod?

MR. BOLIN: No. No.

MEMBER MARCH-LEUBA: So, I thought that was the number that we were given earlier.

MR. BOLIN: The cladding itself. No, it doesn't, it doesn't --

MEMBER PETTI: Decomposition? Because I always thought decomposition was a little higher than

-- around 2000.

MR. BOLIN: Yes, decomposition is higher than that.

MEMBER PETTI: Yes.

MR. BOLIN: It's higher than that.

MEMBER PETTI: So, it's just a composite --

MR. BOLIN: It is degradation. I mean, there's degradation of the cladding at 1800 C. And so, we want to stay below that point. And we are.

MEMBER REMPE: I am confused because, earlier, Aaron said that the peak temperatures for the end caps for air are -- or I guess for helium -- was 1900 C?

MR. BOLIN: Correct.

MEMBER REMPE: So, what are the -- aren't the end caps made of silicon carbide, too?

MR. BOLIN: Yes. Yes.

MEMBER REMPE: Was it 1800 or 1900?

MR. BOLIN: Eighteen hundred was just the number we were using as a design --

MEMBER REMPE: So, you have margin, is what you're saying?

MR. BOLIN: Yes. Yes.

MEMBER REMPE: Unless there's not any

steam in there.

MR. BOLIN: Right.

DR. SCHULTZ: John, this is Steve Schultz. Just thinking about some operational considerations. You mentioned the fuel cycle approach was going to be such that the fuel assemblies would be in reactor for a fairly extended period of operation. Any concerns about the fuel assembly dimensional stability over those long periods of time to high burnups as well? In other words, are you going to have any problems after many, many years moving assemblies when you do your fuel management at infrequent intervals?

MR. BOLIN: Well, we will be moving fuel assemblies every -- so, when we refuel every 15 years, all the fuel assemblies get moved.

DR. SCHULTZ: We hope.

MR. BOLIN: Well, yes. And like I said, 15 years, we will have design gaps around the fuel assemblies. And like I said, silicon carbide is very rigid material. So, no deformation is --

DR. SCHULTZ: So, the accommodation is there in the design --

MR. BOLIN: There is some --

DR. SCHULTZ: -- for that type of an approach?

MR. BOLIN: Correct. Correct.

MEMBER PETTI: How many dpa's, John, about?

MR. BOLIN: Well, on average, it is 77.

MEMBER PETTI: And peak?

MR. BOLIN: No. And peak is about 100. So, pretty aggressive. But it's been done before.

MEMBER PETTI: Yes. No, I'm not too worried about it. There's data that shows that it's okay. I'm more worried about how you qualify a 15-year fuel cycle in ATR, which is like super-accelerated for light water reactors. It's kind of off the charts for this fuel.

MR. BOLIN: It's a subject of a presentation I'm giving on Thursday.

MEMBER PETTI: Oh, good. I'm glad you've got the answer.

DR. SCHULTZ: I am, too. That sounds good.

MR. BOLIN: It's at INL. So, if you're in the neighborhood --

DR. SCHULTZ: Well, thank you, John.

MEMBER REMPE: Back in D.C., though, a question. Since this is your last slide, what are you guys going to do for fuel ultimate storage? Is it going to be onsite? Are there any special concerns about how you're going to store it onsite, and then, ultimately, if we ever have a repository, transferring it to the repository? Or at least --

MR. BOLIN: So --

MEMBER REMPE: Go ahead.

MR. BOLIN: So, we're still working on that. We're going to store onsite within the reactor building, similar to other gas-cooled reactor designs. At least have a core's worth -- I think core and a reload worth of storage onsite in a spent fuel storage area. Because I don't want to say -- I don't know if it's going to be a pool, a vault, or storage wells. That's still being worked out.

Eventually, I think we'll use dry storage on the outside of the reactor building, similar to like what light water reactors are doing now. We may be able to move fuel into dry storage casks fairly early because of our low power density. So, that's still being investigated.

I do have to show my last slide just to acknowledge that this work was supported by the Department of Energy, Office of Nuclear Energy, under that contract for advanced reactor concepts.

So, let's see.

CHAIR BIER: So, we are remarkably close to being on time. You did a good job anticipating how many questions and comments you would get, I guess.

MR. BOLIN: Yes.

CHAIR BIER: If there are no more questions on this presentation, this is probably a good time to take a break, as scheduled, and be back, I guess, at maybe 3:30, instead of 3:25. Is that okay with people? I'm sorry, 2:30. I'm looking at the wrong computer and doing the conversion in my mind of time zones. You're right.

We're ahead of schedule. So, let's take a break until 2:30, and then resume. Is that agreeable to everybody?

(Whereupon, the above-entitled matter went off the record at 2:12 p.m. and resumed at 2:32 p.m.)

CHAIR BIER: All right. Sorry for the brief delay. Are people ready to move forward?

John, I believe you are up after the break, also, for the principal design criteria, is that correct?

MR. BOLIN: Correct. Can you hear me?

CHAIR BIER: Yes.

MR. BOLIN: And is my audio better?

CHAIR BIER: Well, it sounds fine so far.

Okay.

MR. BOLIN: Okay. Because I have donned a set of headphones and microphones. Hopefully, that's going to --

CHAIR BIER: That will probably help, yes.

MR. BOLIN: -- cut back on ambient noise.

CHAIR BIER: Thank you. We appreciate it. And so far, you are not sharing your slides. I believe you're planning to.

MR. BOLIN: No, I've not started sharing.

CHAIR BIER: Okay.

MR. BOLIN: Yes.

CHAIR BIER: That's fine.

MR. BOLIN: Yes. Let me pull up that presentation.

CHAIR BIER: By the way, I thought your slides were quite readable, which is nice. Sometimes they're minuscule eye charts, but these were pretty good. Thank you.

MR. BOLIN: There might have been a few tests, eye tests, on there.

Okay. Let's see here. Okay. How is that?

CHAIR BIER: It looks good. Thank you.

MR. BOLIN: Okay. This is an overview of

the Principal Design Criteria for the Fast Modular Reactor. Okay. There.

So, we did use the NRC guidance in adapting and developing our Principal Design Criteria. The Reg Guide 1.232, as you're aware, developed PDCs for non-light water reactors by modifying and supplementing 10 CFR 50, Appendix A, General Design Criteria. And they did that in three categories: sodium-cooled fast reactor, modular high-temperature gas-cooled reactor, and then, a design-neutral advanced reactor design criteria.

So, we used the ARDC and the MHTGR-DC as starting points, and then, in our Topical Report, we modified the NRC rationale for adaptation of the GDC to our application for the FMR-DCs.

So, I'll just quickly go over some of the key things about the FMR-DCs, and I've organized it in the major categories of the design criteria.

So, the first category is overall requirements, FMR-DC 1 through 5.

So, FMR-DC 1 is the same as the GDC.

Likewise, FMR-DC 2 is also the same as GDC.

FMR-DC 3 is the same as the ARDC.

FMR-DC 4 made a slight change to the

MHTGR-DC. We changed missiles. We expanded it or made it more specific to include missiles originating both inside and outside the reactor helium pressure boundary. And we did that to explicitly cover any missiles generated by the turbine machinery.

And then FMR-DC 5 is the same as the GDC.

The next category is multiple barriers, FMR-DC 10 through 19. The FMR-DC 10 on reactor design, the fuel design, as we went over previously, the fuel design using SiGA cladding, it functions similarly to light water reactors in that, you know, that there's kind of a classic cladding function.

And so, we chose to use the SAFDL terminology, both here and in other FMR-DCs. So, where we may have been using a MHTGR-DC, we, basically, chose to use the SAFDL terminology instead of the -- I don't know how you pronounce it -- SARRDL terminology.

Okay. Then, FMR-DCs 11, 13 through 15, 17 through 18, those are the same as the ARDCs and MHTGR-DCs, but with minor terminology changes. So, they're, essentially, the same as those.

FMR-DC 12, the suppression of reactor power oscillations, the word "structures" was added to address reflectors, but the word "coolant" was deleted.

So, as mentioned, the helium coolant itself is neutronically neutral or inert or transparent. And so, helium density flow changes, in and of itself, don't cause a reactivity change, but --

MEMBER MARCH-LEUBA: This is Jose.

But, conceivably, you could have something like u-tube momentarily-type oscillation of the coolant between the reflector and the core, for example. I don't know if you're supposed to go into that. And that will, even though the helium is (audio interference), it changes the temperature and has some Doppler feedback. I don't suspect, I mean, it's even remotely a problem, but you should analyze it.

MEMBER KIRCHNER: John, before you answer, let me add on.

So, you have a fast reactor here, and you have a reflector in the middle of it. Now, that's an adaptation from the HTGR world. That's to push the power out and to allow your passive heat rejection system to take care of decay heat and manage the vessel wall temperature, and a number of other factors.

But, for a fast reactor now, does that reflector decouple one part of the reactor from another? In other words, do you have a tightly-coupled neutronic design or is this loose now -- I use that word loosely -- in terms of how the core might behave with regard to power oscillations? Because you're now running in a fast spectrum, not a thermal spectrum.

MR. BOLIN: So, I think the coolant, in and of itself, is not a source of power oscillation. But the flow is a possible source of power oscillation. So, I think that's why structures were added to address whether flow could cause the reflectors to move, and therefore, cause a reactivity-generated power oscillation.

Now, that flow through the reflectors can affect both position and temperature. So, I think we're covered by that.

Since it is a fast-spectrum reactor, it should be fairly tightly-coupled, and these reflectors, like I said, this is zirconium silicide. So, it's a fairly heavy reflector. So, it doesn't moderate like a lot of reflectors tend to do, you know, graphite or water, or whatnot.

So, the coupling, I think the coupling is tight, but we're still looking at whether there's really any significant oscillation or reactivity feedback from the reflectors. Of course, historically, Fort Saint Vrain had a power oscillation issue with the fuel columns and their moving around. So, I think that's why we don't want to totally ignore the structures.

Okay? Then, the last DC in this category is containment design. And we are using the same as the SFR-DC because the FMR uses, like we discussed earlier, a low-leakage pressure-retaining containment. So, more in line with the SFR-DC and, certainly, not the vented confinement of the MHTGR.

The reactivity control is FMR-DC 20 through 29.

FMR-DC 20 through 24 are the same as the GDCs.

FMR-DC 25 is the same as ARDC with minor terminology changes.

And then, FMR-DC 26, just like the ARDC and MHTGR-DC, it combines GDC 26 and GDC 27.

FMR-DC 28 is the same as MHTGR-DC.

And FMR-DC 29 is the same as GDC.

Fluid systems, FMR-DC 30 through 46.

30 through 33 are the same as MHTGR-DC.

The FMR-DC 34, residual heat removal, it's similar to MHTGR-DC, but we wanted to make sure that it covered both the active, non-safety-related and passive safety-related systems available to remove residual heat. Also, similar to MHTGR-DC, it incorporates the requirements in GDC 35.

And then, FMR-DC 36 and 37 are the same as MHTGR-DC.

38 through 41 are the same as the ARDC.

DC-42 is the same as GDC.

DC-43, 45, and 46 are the same as ARDC.

And FMR-DC 44 is the same as MHTGR-DC.

So, all of these PDC selections are driven by the design choices that we've made in the design.

And I believe this is the last slide of DCs. It is reactor containment.

So, 50 through 53 are the same as ARDC.

54 is the same as SFR-DC.

And 55 through 57, they're the same as ARDC, but with minor terminology changes.

And then, the next category is fuel and radioactivity control.

60, 62, and 63 are the same as GDC.

And 61 is the same as ARDC.

And the same acknowledgment as the previous presentation. It's supported by the U.S. DOE, Office of Nuclear Energy, under that contract.

So, that was that.

CHAIR BIER: So, this is Vicki Bier. I have a couple of very general, high-level questions.

One, could you discuss briefly, of the modifications you described, which ones were kind of the most crucial for safety versus just matching the terminology to what's in your design?

MR. BOLIN: Well, let's see here.

MEMBER MARCH-LEUBA: How about DC 4 with the missiles?

MR. BOLIN: Yes, that probably is the most unique challenge from the FMR, is the missiles. Because that, obviously, adds -- I mean, not that it wouldn't have been considered, anyway, but it certainly adds a design focus. I mean, not that we would have ignored it, but, yes.

CHAIR BIER: Sure. One other, again, high-level question. I know that the Reg Guide says that it is possible for the applicants to identify entirely new PDCs for unique features of the design that are not adequately covered by the kind of templates in the Reg Guide. And did you find any situations where you were at least pondering that or thought it might be worthwhile? Or do you think you're close enough to the samples in the Reg Guide, that you were able to cover what you needed there?

MR. BOLIN: I think, because the staff covered the two different, very different, advanced reactors -- the sodium fast reactor and the MHTGR -- that I don't think there were -- we did not identify any gaps in design criteria that we -- we did not identify any gaps.

CHAIR BIER: Okay. So, in other words, the reason why what you have looks a lot like the samples in the Reg Guide is really just because the Reg Guide is pretty thorough and comprehensive, not because you were just going through kind of a checkbox process of "pick one from each column" kind of thing?

MR. BOLIN: Correct. Correct.

CHAIR BIER: Okay.

MR. BOLIN: I mean, I think the Reg Guide was extremely useful in this process.

MEMBER REMPE: Well, thinking about the historical approaches that GA has developed, where you start with the critical safety functions, are you doing that or applying that approach with this design? Is it just, you know, control radionuclide release --

MR. BOLIN: It's the same critical safety functions, correct.

MEMBER REMPE: So, there's nothing that's unique or different, then, when you think about this design? I mean, sometimes chemical reactions comes in with higher priority, but you just didn't see anything else?

MR. BOLIN: No. I mean, obviously, we still have some graphite. So, we do have graphite concerns. But we don't have -- the graphite is not in the high-temperature parts of the core.

We still have water ingress concerns, but we don't have high-temperature, high-pressure steam. So, a lot of our safety concerns from MHTGR are quite a bit lessened.

MEMBER MARCH-LEUBA: This is Jose.

How about the very long cycle time, the 15-years recycle/reload, and the implications that you may have on misalignment of fuel, clipping, phase-in, moving, vibrations? And 15 is a long time before you open and look inside to see what's going on.

MR. BOLIN: Yes.

MEMBER MARCH-LEUBA: I mean, does that affect something?

MR. BOLIN: Certainly, we do expect to have to shut down more frequently than every 15 years for other maintenance and inspection reasons.

MEMBER MARCH-LEUBA: Yes, but do you

expect to open the core? You'll probably be fixing some pump outside of the core plenum.

MR. BOLIN: Well, that hasn't been decided, whether --

MEMBER MARCH-LEUBA: Yes. And if I incorrectly wanted to --

MR. BOLIN: And, you know, for the first-of-a-kind prototype, it might be you might do some fuel inspection.

MEMBER MARCH-LEUBA: I'm just trying to think what is different. If I read correctly your cartoons, the control rods are sitting outside, or the shutdown rods for sure are sitting outside the vessel, and they have to go through sealed?

MR. BOLIN: Well, the control rod drive mechanism and connecting rod are out -- well, they're not -- technically, that's still part of the vessel. It's still part of the helium reactor pressure boundary.

MEMBER MARCH-LEUBA: Then, enclosed? I'm just wondering if there is some design criteria that applies to those special configurations.

MR. BOLIN: Well --

MEMBER MARCH-LEUBA: It certainly feels -- let me put it this way; I'm the bad guy here -- it simply feels that you took all the GDCs that were in the design guide and went through to see if they applied to you, instead of thinking about your design and see what's missing. It's something very human to do, and that's something we all do.

So, on the review, I'll be asking the staff, when they're here, if they thought, what's missing?

MR. BOLIN: Okay.

MEMBER MARCH-LEUBA: It's very easy, when somebody gives you a paper, to correct the English, but what's important is, what paragraph is missing in that article? The same thing here.

MR. BOLIN: All right.

CHAIR BIER: For operational reasons, are you anticipating that there would be periodic shutdowns for reasons other than refueling, or that it's just going to run flat-out and just adjust power levels?

MR. BOLIN: I mean, it hasn't been worked out specifically, but there is discussion of whether we would want to shut down every five years for an inspection, particularly, you know, power conversion unit inspection and/or generator or control rod drive motors, or a variety of things we might want to inspect every five years. At least, particularly, we have to consider increased inspection frequency for a first-of-a-kind plant. So, that's being discussed.

MEMBER PETTI: Wouldn't Section 11 require, like, the vessel to be inspected?

MR. BOLIN: Yes, vessel inspection is another example.

MEMBER HALNON: Yes, and don't underestimate the power of the insurance agency.

MR. BOLIN: Well, yes. We tend to ignore that until the very end.

CHAIR BIER: Are there other questions or comments for John, or any other points that John wants to add, before we transition to the staff?

(No response.)

CHAIR BIER: I guess one other question that I have, I noticed that there was a Rev 1 of the PDCs, which I guess was in response to the RAIs from the staff; that some things got adjusted? Again, are there any there that are noteworthy enough that you want to call them out or discuss the value of those changes?

MR. BOLIN: Well, it's interesting that a lot of the changes were -- we had actually prepared our DCs based on a draft of the Reg Guide. And then, the Reg Guide changed. And so, there were inconsistencies between our DCs and the Reg Guide. So, a lot of the corrections were just making those corrections to the revised Reg Guide.

CHAIR BIER: Okay. If there are no further questions and comments, then, I guess we can transition to staff. And I'm not sure if the primary presenter is Reed Anzalone or Samuel Cuadrado. Which of --

MR. ANZALONE: It's going to be me.

CHAIR BIER: Okay. Thank you.

So, I guess, John, you can stop sharing your slides, then. Thank you very much for the presentation.

MR. BOLIN: Thank you.

MR. ANZALONE: Okay. Thanks. So, thank you, everyone, for having us here today.

My name is Reed Anzalone. I'm a Senior Nuclear Engineer in NRR's Division of Advanced Reactors and Non-Power Production and Utilization Facilities. I'm joined today by our Project Manager, Sam Cuadrado, who is also with me in DANU.

So, I was the lead technical reviewer for this effort, and I was assisted by Sheila Ray in our Division of Engineering and External Hazards, who's here on the phone. She covered the electrical PDCs, and Steve Jones, who is with me in DANU, covered the containment PDCs, who wasn't able to make it. So, if there are questions about those, I can address them.

Next slide, please.

So, quick agenda, and you'll see that a lot of this should look very, very, very, very familiar from the presentations that we just had from General Atomics. I was laughing the whole time during John's presentation because there is almost a one-to-one correspondence between the topics covered. So, I may go quickly through some of these. And, of course, if you have questions, feel free to interrupt.

MEMBER MARCH-LEUBA: Is that the 100 percent rule to coordinate with the other presenter?

MR. ANZALONE: No. In fact, we --

MEMBER MARCH-LEUBA: In fact, the desirable?

(Laughter.)

MR. ANZALONE: We only got the slides yesterday. So, I was happy to see that they matched very well.

So, Sam will be talking a little bit about the pre-application engagement. Then, it will go back to me, and I'll talk about the Topical Report timeline. That's really just going to be for reference, for anyone who might want to go back and look at the correspondence that we had.

I'll touch a little bit on some of the design features that we already talked about; talk a little bit about the PDC guidance that's out there; the PDC development approach that General Atomics provided to us in their Topical Report, and then, I'll go into the Fast Modular Reactor design criteria themselves, including kind of highlighting the key design choices and the effects that those had on the PDCs. And hopefully, I can address the question that you raised. And then, I'll just briefly touch on the Safety Evaluation and conclusions.

So, Next slide. MR. CUADRADO DE JESUS: Now, good afternoon. Sam Cuadrado.

So, this is a brief overview of the pre-application engagement with General Atomics. You saw a similar slide when John Bolin was doing the presentation.

We got the pre-application letter engagement plan last year in March. Accordingly, we're reviewing a couple of documents, which include this one that we see, the Topical Report which is the topic of this meeting.

We have a few qualifications in the Topical Report. Basically, it's only that we are providing feedback in the form of a white paper. And last month, we received the Quality Assurance Program Topical Report. That's currently going through the review process.

We expect a few more documents, a few more submittals this year, and a couple more next year. This year, for the summer, we've got the mechanistic source term. By the end of the year, we should be getting the licensing basis event white paper. And for the spring of next year, the safety approach on the PRA and safety classification white papers.

So, back to Reed on this.

MR. ANZALONE: All right. Next slide. So, just quickly on the review timeline, really, all I wanted to highlight here was that we did ask a round of RAIs and we got prompt responses, and then, subsequently, General Atomics rev'd the Topical Report to incorporate those responses. And then we issued the Draft SA in March.

Next slide. Well, go ahead.

MEMBER BALLINGER: I have a question. I see that the Fuel Qualification Plan Technical Report, it says, "under review." Do we know when we're going to get that?

MR. CUADRADO DE JESUS: Yes. That is for a white paper there. It was just some feedback. So, they provided some "asks," some questions for us to provide them feedback. I placed that information for you guys to get access to it. But we plan to provide feedback to them by November of this year.

MEMBER BALLINGER: So, we have access to this?

MR. CUADRADO DE JESUS: You have access to the request, yes, to the Technical Report.

MS. DE MESSIERES: This is Candace de Messieres.

I just wanted to clarify that this is a Technical Report, not a Topical Report.

MR. CUADRADO DE JESUS: Yes, yes, yes.

MS. DE MESSIERES: So, I just wanted to make sure that that was clear.

MR. CUADRADO DE JESUS: Yes, but you can see the Technical Report and the questions that they want us to answer, to provide feedback. It's in SharePoint for you guys.

MEMBER MARCH-LEUBA: But it has been provided on the docket? I mean, we can see it?

MR. CUADRADO DE JESUS: Yes, it's on the

docket.

MEMBER MARCH-LEUBA: But, typically, Technical Reports are part of the SAR.

MEMBER HALNON: Yes, we, typically, don't -- we separate white papers from Technical Reports. This has both. Is it a white paper or is it an actually approved -- since they don't have a QA program yet, it can't be a Technical Report that they would reference.

MR. ANZALONE: No, it's a white paper.

MEMBER HALNON: Okay.

MR. ANZALONE: It's just called a Technical Report.

MEMBER HALNON: Yes, that's the title.

(Laughter.)

MR. CUADRADO DE JESUS: Yes, but it's on the docket, so you guys can see it.

MR. ANZALONE: So, it's not going to be, you know -- it doesn't get that stamp of finality that's --

MEMBER HALNON: It won't be referenced out of the SAR --

MR. ANZALONE: No.

MEMBER HALNON: -- whenever that comes.

CHAIR BIER: Just for completeness or

clarification, Weidong, if I understand correctly, this is not in the SharePoint for this meeting, but it's available through NRC, is that correct?

MR. WANG: Correct. That is, the staff has created a SharePoint, but I think that Sam has --

MR. CUADRADO DE JESUS: Yes, in SharePoint, there's a folder related to General Atomics.

MEMBER HALNON: So, on your next slide, Rev 2 was transmitted. They make it Rev 1.

MR. ANZALONE: I think it should be Rev 1. Sorry.

Okay. Next slide. So, just talking a little bit about the design features, I know we've just, literally, had a presentation from them. I just wanted to kind of go through the things that we thought were particularly noteworthy in our review.

So, one is, obviously, the core arrangement, which is different from the other gas-cooled reactors that we've seen recently at the NRC, which, you know, they're using, essentially, what looks like an LWR core with the fuel rods and UO2 cladding, the silicon carbide. But the arrangement is a little bit more like a fast reactor core with a tight space in between the rods and triangular tension, the hexagonal assemblies.

And when you compare that -- so, I was thinking about things in terms of the basis for what's in Reg Guide 1.232. So, that's the MHTGR, which was a prismatic block gas-cooled reactor using TRISO fuel. So, obviously, pretty different there.

The other thing that's big that you can see on this slide is the gas turbine. I think that's been covered pretty well.

And then, the MHTGR used steam generators rather than having the power conversion system directly on the primary circuit.

Next slide. The thing that I'll highlight here, you see the containment. John talked a little bit about that. So, there is an actual containment building versus like a functional containment or confinement approach.

And the other thing is the RVCS cooling system, which is a little different from the passive cooling system that was in the MHTGR design.

Yes, that's everything I wanted to cover on this slide.

So, then, the key design features, and on a future slide, I'll kind of talk about how these feed into the PDCs. So, it just sort of sums up everything that I covered in the last couple of slides. I don't think I need to really talk any more about this.

Next slide. MEMBER KIRCHNER: Can you just pull your microphone up? Maybe we can hear it better. Yes, great.

MR. ANZALONE: All right. Sorry. Thanks. Oh, that is much louder.

MEMBER MARCH-LEUBA: There's people on the other side of the phone line that would love to see how this works.

MR. ANZALONE: Yes. Thanks. That's much better.

So, just a little bit of what we used for guidance in evaluating PDCs and you know the kind of conclusions that we're trying to reach. So, both of these quotes are from Part 50, Appendix A.

And that first one, the first statement there is kind of the conclusion that we're trying to reach: that the Principal Design Criteria established the necessary design, fabrication, construction, testing, and performance requirements.

And then, the second statement there talks about this is the guidance that Part 50, Appendix A, gives in establishing PDCs. So, the GDCs in Appendix A aren't directly applicable to non-light water reactors, but they are considered to be guidance in establishing PDCs for non-light water reactors.

Next slide. And then, also, we have Reg Guide 1.232, which we talked about a little bit. That was issued in April of 2018. I think most of the members were on the Committee when it was issued.

And it documents three sets of acceptable PDCs. So, there's the advanced reactor DCs, which are supposed to be generic and technology-inclusive, and there's an asterisk there because it's technology-inclusive for certain technologies that we had in mind when we were writing them. I don't think that you could make one that is, you know, wholly generic that would be of value really.

Then, there is the sodium-cooled fast reactor DC, which really, I think, were made with the PRISM reactor in mind, and the MHTGR-DCs, which were made with the MHTGR which is a TRISO-fueled, helium-cooled, as I mentioned, prismatic block, graphite-moderated, high-temperature gas reactor.

So, that slide --

CHAIR BIER: Before you move on, when these design criteria were developed, did you envision in advance that people might be mixing and matching, based on what fits their circumstance?

MR. ANZALONE: Yes. And actually, there's one -- one of the points coming up, especially PDC 16 talks about containment design criteria. That explicitly in it says, "We envisioned that people would pick the one that best suits their design here." So, I think that was, clearly, a consideration.

And the thing that I want to kind of point out -- and this gets a little bit at Jose's question -- the FMR kind of neatly straddles all of these categories. It falls kind of in between all of them. So, I don't think that there's any real aspect of the design that is so exotic that it wouldn't be well-encompassed by these design criteria.

And then, that is something that we thought about, as we were going through and doing the review, is, you know, are these adequate? And the answer that we keep coming back to was, yes, it looks like this covers what it needs to.

MEMBER MARCH-LEUBA: And an impracticality -- if that's a word -- when we do the full Chapter 15-type analysis and Chapter 19, it will mean something. It will pop up there.

MR. ANZALONE: Mm-hmm. And one thing that I think is interesting about this particular review is that they came to us with these PDCs very, very early in the project. I mean, this was the second thing that was submitted after the Regulatory Engagement Plan. So, really, it's the PDCs came, and then, the design -- I mean, it got to a certain level of maturity to be able to establish what the PDCs ought to be, but, ultimately, they will have to design the reactor to meet these PDCs.

MEMBER HALNON: But is it, I mean, written generically enough to create a fourth category in regards to the Reg Guide?

(Laughter.)

MEMBER HALNON: I mean, when I went through it, it seemed like there were pretty generic, directly written to advanced reactors of this type.

MR. ANZALONE: So, I don't know that that's really necessarily worth doing. Because I think there's this vision that people would kind of mix and match. The vast majority -- and I have a summary slide as a backup slide -- the vast majority are just straight from the ARDC with minor modifications here and there.

MEMBER HALNON: Okay.

MR. ANZALONE: So, I think it's, you know, within the envelope of stuff that we would expect people to do with this Reg Guide.

CHAIR BIER: And presumably, any future designs may also have their own unique tweaks and not fit exactly with --

MEMBER HALNON: But when I got done reading it, one of the things that just popped into my mind, it just felt like the fourth category was just written, but I understand what you're saying. They're close enough to all these other things.

MR. ANZALONE: Yes.

MEMBER HALNON: There's nothing really unique or brand-new in there that would warrant a special --

MEMBER MARCH-LEUBA: If anything worries me along this design, it's the high temperature. But they'll eventually know how to do it.

MR. ANZALONE: All right. Next slide. And General Atomics covered this in their presentation, but the concept that was conveyed to us in the Topical Report was that they would start with the Advanced Reactor Design Criteria. Then, if that wasn't fully applicable, they would go look at the other ones for direct adoption, and then, take the one that was the most applicable and adapt or refine it to match with the design.

Okay. Next slide. So, then, talking

about the key design feature effects on the Principal Design Criteria. So, the fuel and the core really kind of lead to the use of SAFDLs rather than SARRDLs, which John mentioned in his presentation. And that also, I think, goes along with the Containment Principal Design Criteria that they ended up using and the containment design. We generally kind of think of those as going together. SAFDLs go with functional containment; SARRDLs go with leak-tight containment or controlled leakage.

The neutron spectrum fast, I think the big takeaway there that we wanted to make sure was included was to consider the effect of structures on reactivity feedback, which, otherwise, I think the ARDC includes this, but the GDC, if that were to be adopted directly, does not.

And actually, that was an RAI that we asked. Because, originally, what was in the Topical Report didn't include structures in that PDC, and we wanted to make sure that that was in there.

The helium coolant, so for that, the big thing there was -- they're all out of order on my paper here -- that affects a whole bunch of the Principal Design Criteria. The big effect is to remove considerations related to coolant inventory control, and that's consistent with the modular high-temperature gas reactor design criteria. And that means there's no PDC 35, which relates to emergency core cooling systems. And the emphasis is placed on the residual heat removal systems.

And then there's also they decided to change, to be consistent, the reactor helium pressure boundary, instead of reactor coolant pressure boundary or reactor coolant boundary in the PDCs.

There wasn't any particular effect for the gas turbine on the primary coolant. DC 4 has the consideration of missiles generated from either inside or outside the containment. That was actually included originally in the MHTGR-DC 4. So, I wanted to note that there.

The residual heat removal -- and John mentioned this on his slides -- so, they adopted the MHTGR passive residual heat removal PDCs, but, then, they adapted them to remove passive, so that it would encompass both their passive system and the active non-safety-related system.

And that seemed appropriate to us to do. You know, it just made it broader, so that it covers a wider scope of systems; whereas, the MHTGR-DC really just, specifically, covers the passive residual heat removal systems.

And then, for containment, I already mentioned that they got the leak-tight containment building, and they adopted the standard containment PDCs.

Next slide. So, I've, basically, already covered most of these in what I just said, but we can quickly go through this.

So, I listed out all of the design criteria, and then, highlighted ones that I think are worth mentioning, either because they had to make a particular choice about where they went with it or they've modified it in an interesting way.

So, I just finished talking about PDC 4. It's noteworthy because they wanted to include missiles generated inside the reactor helium pressure boundary.

Next slide. So, 10, we've got the --

MEMBER BROWN: Can I ask you a question about the missiles?

MR. ANZALONE: Sure.

MEMBER BROWN: They show in their prior generation a gas turbine-driven, you know, the heated helium-driven TGs. Is that a very high speed? Is that a very, very high-speed? I mean, all plants have a missile issue relative to their turbine generator sets you have to consider. So, this looked like a high-speed one, which would make it more critical in terms of covering it. Is that the reason for the emphasis here?

MR. ANZALONE: No. I think the reason is just that, like, it is noteworthy that they have a power conversion system that is inside containment.

MEMBER BROWN: Oh, okay. All right.

MR. ANZALONE: And it's part of the reactor coolant boundary.

MEMBER KIRCHNER: It's part of the primary cooling boundary.

MEMBER BROWN: Yes. Okay. So, that part I missed. I missed that it was inside the coolant, the primary boundary.

MR. ANZALONE: Yes.

MEMBER BROWN: My brain fried on it.

(Laughter.)

MEMBER BROWN: Thank you.

MR. ANZALONE: Go ahead.

So, I already mentioned the use of SAFDLs rather than SARRDLs. I will also mention that Criterion 10 uses, talks about heat removal, rather than coolant. So, I think the effect that you were talking about of, you know, potentially, oscillatory coolant behavior -- I think that talking about heat removal is appropriate there.

The same thing with we talked about power oscillations. Coolant isn't mentioned there, but it talks about the core. And so, I think that's appropriately considered, enveloped by that design criterion. And it does have structures in there in talking about power oscillations.

So, if there's an effect of the reflectors, that's covered under that design criterion. Whether that effect is caused by the behavior of coolant affecting the structures or it's something in the inherent behavior of the structures.

MEMBER MARCH-LEUBA: Yes, I sense some real thinking that you worry about the structures because of mechanical vibrations or displacement; whereas, it could be a temperature oscillation. I find it very unlikely that will happen, but --

MR. ANZALONE: Yes.

MEMBER MARCH-LEUBA: -- you have to consider that it will happen.

MR. ANZALONE: No, but, I mean, actually, I will say, part of the reason that we asked about structures was, you know, knowing fast reactors and that they're, basically, a coupled system, we felt like that was important to include. We didn't know at the time when we asked that question that all of the structures are going to be made out of silicon carbide. And so, they don't really move very much during power maneuver.

MEMBER MARCH-LEUBA: But you have those unusual bowing effects. It's an oscillation configuration, a crucial thing.

MR. ANZALONE: But we think it's covered by just making sure that structures are considered in there.

MEMBER MARCH-LEUBA: Okay.

MR. ANZALONE: Thirteen, that was one where they used the helium pressure boundary instead of the reactor coolant boundary. And I know it's instrumentation and control, but, really, the main distinguishing feature between all the different DCs was what the coolant system looked like.

Containment design. John already covered that, I think in sufficient detail.

And electric power systems, they used the MHTGR design criterion, but they modified it to go with SAFDLs instead of SARRDLs. That's appropriate to be consistent with the other DCs.

MEMBER MARCH-LEUBA: Do we envision safety-grade power? Or there is nothing that needs to be driven?

MR. ANZALONE: I can't remember off the top of my head. I'm going to phone a friend.

(Laughter.)

MR. ANZALONE: To Sheila, do you think you can answer that question? Or John?

MR. BOLIN: John can answer it.

We don't see a need for Class 1E backup electrical generation.

Does that --

MEMBER MARCH-LEUBA: Yes. But you usually have a couple of batteries for the control room, right?

MR. BOLIN: Correct. Correct. We'll have --

MEMBER MARCH-LEUBA: Non-safety grade?

MR. BOLIN: Or, you know, there will be containment isolation valves, other isolation valves that might need to function. Whether they're electrical, by battery, or some other means is still to be determined.

MR. ANZALONE: Next slide. So, 26. I want to highlight here we have had some challenges with some applicants in PDC 26. I'm not going to go into that in detail here. But here, they adopted the PDC, the language in advanced reactor design criteria.

As is, with one modification, to be consistent with the GDC, they included the effects of xenon. We don't think that xenon is going to be particularly important in a fast reactor, but there wasn't any reason not to include it.

MR. BOLIN: And I'll second that. We have recently found that also to be the case, that xenon is really not of any -- it has no impact to speak of.

MR. ANZALONE: But it's included in the design criterion.

MR. BOLIN: But it's there. It's there.

MR. ANZALONE: So, if somehow it's found to have an effect, it's covered. More broad is actually okay.

So, consistent with the all of the sets of design criteria in the Reg Guide, got rid of PDC 27 and incorporated it into 26.

MEMBER MARCH-LEUBA: Now that I see, I'm asking not PDC, but criteria limits. Is there an issue with rod ejection here? We have 7 megapascal. Will that be a licensing basis event?

MR. ANZALONE: I would think so.

MEMBER MARCH-LEUBA: Okay.

MR. ANZALONE: We haven't looked at the -- we haven't gotten in the level of detail of understanding the control rod design or the control rod drive systems.

MEMBER MARCH-LEUBA: It's likely my lack of familiarity with fast reactors, but even events on fast reactors bothers me a lot.

MEMBER KIRCHNER: It's a high-pressure envelope and the control rod mechanism is part of the envelope, inside the envelope. So, it's the same as a PWR when it comes to rod ejection.

MEMBER MARCH-LEUBA: Mm-hmm.

MR. ANZALONE: But that is a distinction to sodium fast reactors which are not high-pressure.

For the reactivity limits, they went with the modular high-temperature gas reactor design criterion because it fit the best with the coolant system design that they have. Again, that was the biggest distinguishing feature between all of them.

Next slide. So, getting into fuel systems, I think John already -- I think we covered this. So, 33 and 35 were removed, and that's consistent with the modular high-temperature gas reactor design criteria, and like I said, reflects a focus away from coolant inventory control and towards residual heat removal. And then, those residual heat removal PDCs were adjusted to not specifically mention passive systems.

Next slide. There wasn't anything, in particular, that I wanted to highlight about these, but I saw that, on John's slides, he mentioned that containment heat removal, cleanups, and events were things that they were considering. We think that those are encompassed by these design criteria, the way that they're written.

Next slide. So, here, 54 I think is interesting because they referenced the sodium fast reactor design criteria. They made a change to it to remove reactor, to signify that there are more structures inside. So, normally, it says, "reactor containment," but they want to say, hey, we've got a lot inside containment, aside from just the reactor. The power conversion system is inside containment. So, it just says, "containment," rather than "reactor containment."

But, aside from that, the interesting thing about SFR-DC 54 is that it talks about not necessarily having to isolate systems that penetrate containment where you wouldn't expect a release path.

And that's key for them to be able to have the water tanks for the RVCS outside of containment with lines that go into containment. And because you wouldn't expect a release pathway to go through those pipes, that's acceptable. And that's consistent with the SFR-DC.

MEMBER MARCH-LEUBA: And with 55, the helium pressure boundary doesn't cross containment, does it?

MR. ANZALONE: No, it does not.

MEMBER MARCH-LEUBA: I mean, there might be some feedline.

MR. ANZALONE: Yes.

MEMBER MARCH-LEUBA: But that wouldn't enter part of the pressure containment?

MR. BOLIN: I will correct. There is a system that has helium in it that is connected to the pressure boundary that does cross the containment boundary, and that's the helium purification system.

MR. ANZALONE: Right.

MEMBER MARCH-LEUBA: Yes, but that would be a small line.

MR. ANZALONE: Yes.

MR. BOLIN: It will be a small line, and it certainly will have isolation valves on it.

MR. ANZALONE: Well, and that's what the design criterion says, that you need to do this when you have reactor helium systems penetrating it.

MR. BOLIN: And we will do that.

MR. ANZALONE: And then, the next slide. So, nothing particular with these. They all adopted the advanced reactor design criterion as written.

Next slide. So, I just wanted to talk briefly about the conclusions and the Safety Evaluation. So, we think that they, appropriately, considered the Reg Guide and developed a sufficient set of PDCs that were appropriate for establishing the requirements for the FRM design. And like I said, they came early. So, these will be criteria that we'll open to, as they continue interactions with us.

And what the SE says is that they establish the necessary design, fabrication, construction, et cetera, that 10 CRF 50, Appendix 50, kind of establishes as the requirement for Principal Design Criteria.

And then, I wanted to make note that the Topical Report can be used by future applicants for the FRM, but the way that we do Topical Reports, you know, you have to justify the applicability of the Topical Report when you come in and you use it. And so, there isn't a specific limitation and condition that says it has to be like this reactor. But we expect that, if somebody were to use this Topical Report and reference it, they would have to justify that if it was substantially different in any way, why it was okay to use this design.

MEMBER MARCH-LEUBA: By "somebody," do you mean like a different company?

MR. ANZALONE: Well, presumably -- so, it's for the FRM design. So, I don't know -- some other company bought that design from General Atomics or if they spun off a subsidiary or --

MEMBER MARCH-LEUBA: Doesn't GA own the intellectual property on the Topical Report? I mean, nobody can use it without GA's permission. Well, it's static.

MR. ANZALONE: And that's my last slide. I have some backup slides that go over some of these in more detail.

CHAIR BIER: So, again, a different version of the same question that I asked John earlier with regard to the RAIs. How many of those subsequent changes to Rev 1 were because the Reg Guide itself changed? How many were kind of minor editorial improvements? And were there any that you thought were really (audio interference)?

MR. ANZALONE: I don't think there were any that specifically -- so, the one that I mentioned earlier, which was including structures in the power oscillations, that was an RAI, and that was, we think, important to capture something that was missing.

I think most of the rest of them were, hey, you said you're using this design criterion from the Reg Guide, but the words that you're using don't match up. And that could reflect what John said, that they were using the draft version of the Reg Guide. I think that covers pretty much all the RAIs between those two.

CHAIR BIER: Are there any other questions for Reed and Sam? Sorry. Yes, are there other questions for Reed and Sam? Are there in the room or online?

MEMBER KIRCHNER: I'd just make an observation or two.

I mean, you asked both the applicant and the staff -- my take, the major thing that's different here is that, you know, this concept is straddling the MHTGR and the fast, as has been pointed out, the fast reactor PDCs and the MHTGR. So, the MHTGR, as we know, is a functional containment approach -- SARRDLs, if I got the acronym right, where this is SAFDLs and a containment.

And I think the main issue that I see coming is where the systems straddle the containment boundary for this reactor. So, you've got quasi-passive systems, if I could call them that. The MHTGR was meant to be a passive decay heat removal. Here, you've got a combination of passive/active/quasi-active systems, maybe depending on what design approaches we see presented, but you would really hear about containment bypass, which is not in the MHTGR designs. That, essentially, is confinement and reliance, mainly, on the functional containment. So, I think that's interesting from my vantage point, looking at how they picked and choose, and how you reviewed their use of the Reg Guide.

That would be the two areas I would zero-in on for this particular design. And it begs the question, like Charlie was asking, you know, which systems are active in terms of which ones might need electric power, or will they fail safe, so to speak, without power?

MEMBER PETTI: Well, it just seems to me that there's sort of a body of knowledge of MHTGR and there's a body of knowledge to fast sodium systems.

And this kind of sort of puts them together. There's an intersection, if you will. So, when one thinks about accident response, you know, there may be something there that, when you look at them in accident space, something new comes up that you wouldn't necessarily see looking at them each separately. So, it's just something that, when you get into the details, you have to be looking for.

CHAIR BIER: Additional questions or comments from members or consultants? Anybody on the line?

If not, then we are going to take comments from the public somewhat earlier than is indicated on the agenda.

We'll wait another 30 seconds or so, in case anybody is trying to unmute.

(No response.)

CHAIR BIER: Okay. It sounds like we have no public comments for today.

So, at this point, we have time for member discussion. And I forget if that should be public or not public.

MEMBER REMPE: We can go off the record until regular order. But hope someone will show up tomorrow at 8:30 for us.

CHAIR BIER: Okay, you got that message, Court Reporter? I don't believe there is a need for a closed session. So we will see you at 8:30 in the morning tomorrow, or whoever it is. Thank you.

(Whereupon, the above-entitled matter went off the record at 3:39 p.m.)

General Atomics Electromagnetic Systems

Fast Modular Reactor Conceptual Design

May 2, 2023

Presented To: Advisory Committee on Reactor Safeguards

Prepared By: John Bolin (GA-EMS)

Fast Modular Reactor (FMR) Distinguished Design Team Goal

The goal is to advance General Atomics Electromagnetic Systems (GA-EMS) Fast Modular Reactor 44 MWe (FMR) for flexible and dispatchable carbon-free electricity generation, targeting commercial operations by 2035.

- The Team will develop and verify key-design attributes such as fast-spectrum neutron, inert helium gas coolant, pellet-loaded fuel rod, site-flexible, small and passive heat removal systems, that will result in a safe, maintainable, and cost-effective nuclear power generation.
- The FMR project was officially awarded on 15 December 2021 as a three-year program under Advanced Reactor Demonstration Program (ARDP)/Advanced Reactor Concepts-20 (ARC-20).

GA-EMS is working with a distinguished team to this effort with a strategic partner in Framatome (FRA) as well as collaboration with Electric Power Research Institute (EPRI), Idaho National Laboratory (INL), Argonne National Laboratory (ANL), Sandia National Laboratories (SNL), University of Wisconsin-Madison (UWM), and University of Texas-Arlington (UTA).

Project Objectives That Enable Future Development and Deployment

The project objective is the FMR conceptual design with verifications of key metrics in fuel, safety, and operational performance.

- The project will verify the technical feasibilities of high-burnup fuel operation, passive safety features, and rapid grid adaptability.
- The project includes pre-application licensing activities with Nuclear Regulatory Commission (NRC).
- The project will conduct an initial cost evaluation.

Experimental verification:

(i) Fuel fabrication campaign, high-burnup irradiation tests and transient tests to qualify the fuel design.
(ii) Scaled tests of the Reactor Vessel Cooling System (RVCS) to verify the passive cooling capability.

Numerical verification:

(i) Accident analysis to support the design work and the pre-application licensing.
(ii) FMR plant simulator to demonstrate the rapid load-following capability.

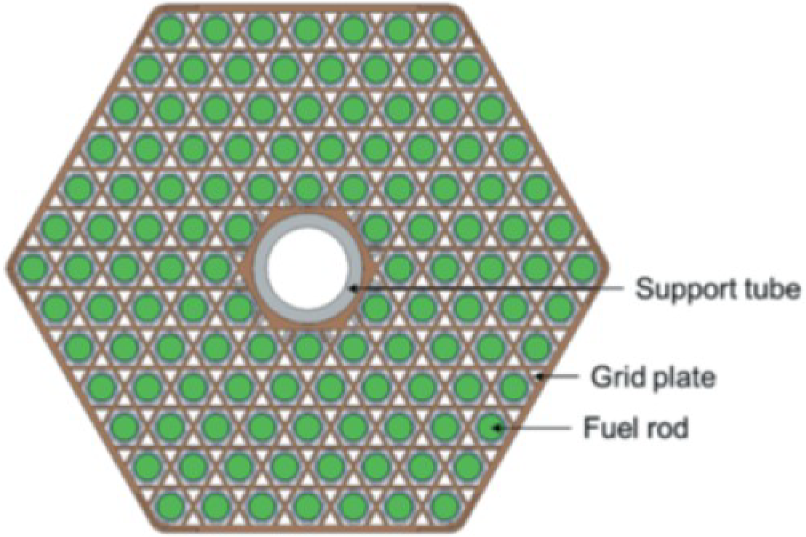
FMR Core Designed to Improve Safety Margin

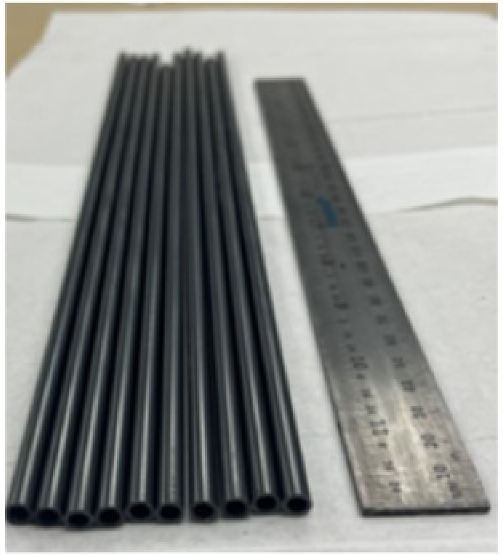
	FMR	GT-MHR	AP1000
Reactor core heat output, MWt	100	600	3400
Reactor core power density, MW/m^3	14.97	6.6	109.7
Heat generated in fuel, %	-	-	97.4
Nominal system pressure, MPa	7	7.07	15.5
Coolant total flow rate, kg/s	66	320	14,301
Coolant nominal inlet temperature, °C	509	491	279.4
Coolant temperature rise in core, °C	291	359	27.4
Fuel rod average linear power, kW/m	**2.34**	**0.39**[a]	**18.8**
Heat flux hot channel factor, F_Q	**1.52**	-	**2.6**
Fuel assembly geometry	Hexagonal	Hexagonal[b]	Square[c]
Number of fuel assemblies	198	102[d]	157
Fuel rods per assembly	120	210[a]	264
Fuel material	UO_2	$UC_{0.5}O_{1.5}$	UO_2
Cladding material	SiGA	SiC[e]	ZIRLO
Core active height (cm)	180	793	426.72

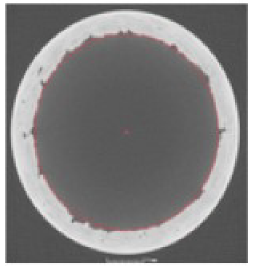
a) Stack of fuel compacts, b) Solid block with coolant channels inside, c) 17×17, d) Fuel blocks, e) TRISO fuel particle coating

Fuel Leverages UO_2 Legacy and SiGA™ Cladding Development

- High density UO_2 proven in LWRs and tested in fast reactors
- Silicon carbide composite cladding (SiGA) undergoing testing and maturation through DOE Accident Tolerant Fuel (ATF) program
- Fuel design uses ATF-LWR dimensions with large plenum like legacy liquid metal fast reactors

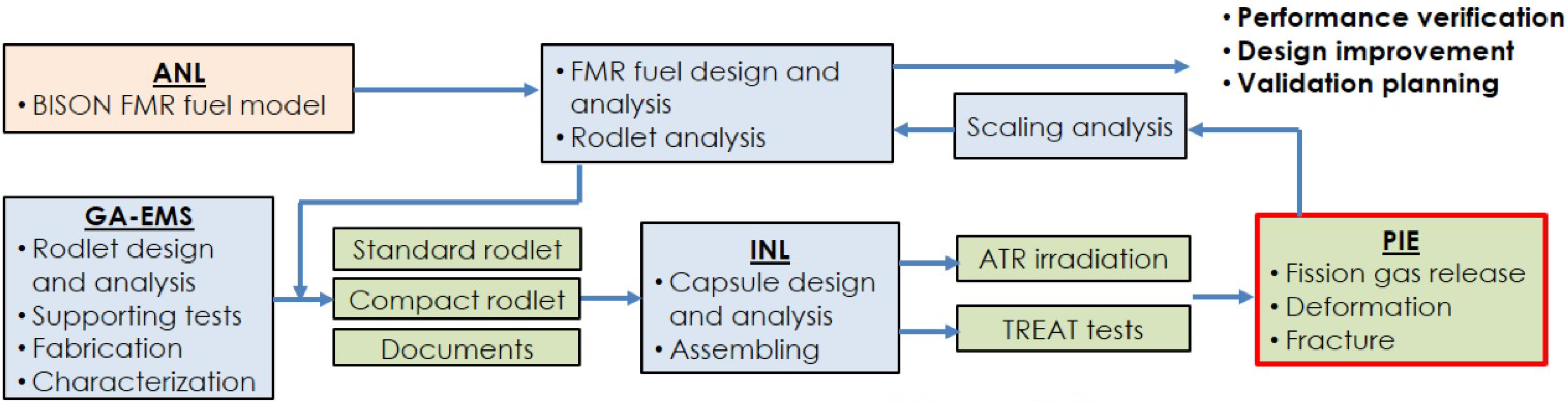





Cladding tube


Endcap welding

Numerical and Experimental Verification of Fuel Design



ANL
- BISON FMR fuel model

- FMR fuel design and analysis
- Rodlet analysis

- **Performance verification**
- **Design improvement**
- **Validation planning**

Scaling analysis

GA-EMS
- Rodlet design and analysis
- Supporting tests
- Fabrication
- Characterization

Standard rodlet

Compact rodlet

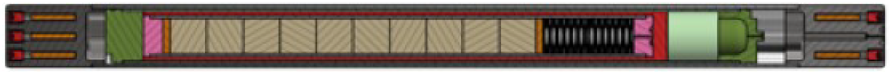
Documents

INL
- Capsule design and analysis
- Assembling

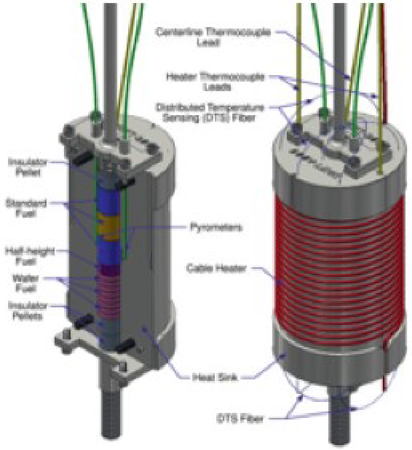
ATR irradiation

TREAT tests

PIE
- Fission gas release
- Deformation
- Fracture



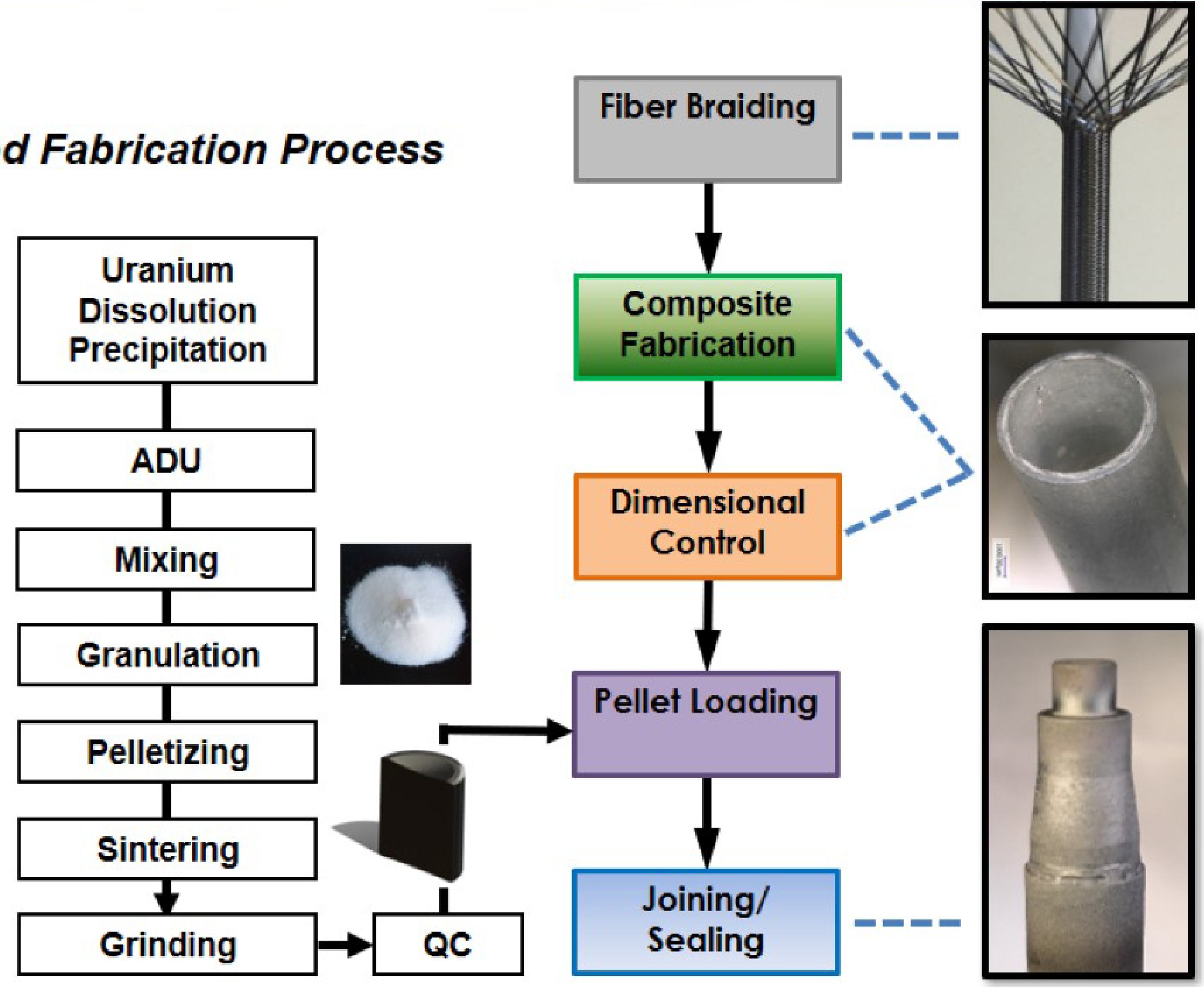
ATR Irradiation Capsule



TREAT Transient Capsule

FMR Test Rodlets Fabricated Using ATF Established Procedures

Fuel Rod Fabrication Process

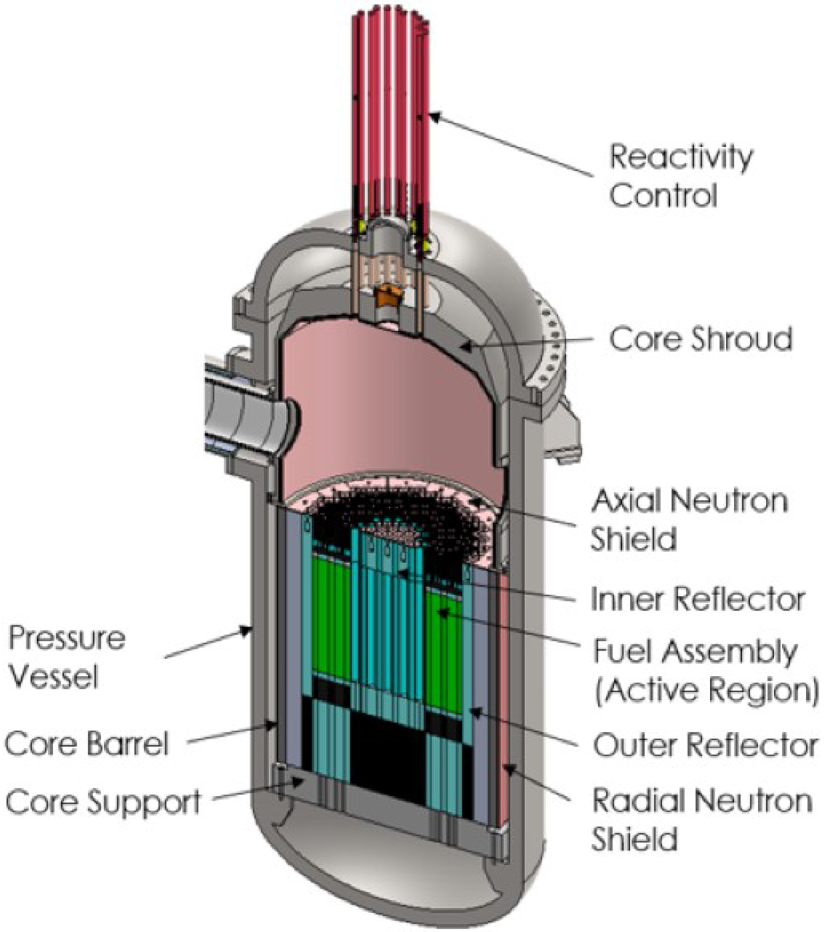


Uranium Dissolution Precipitation
ADU
Mixing
Granulation
Pelletizing
Sintering
Grinding
QC

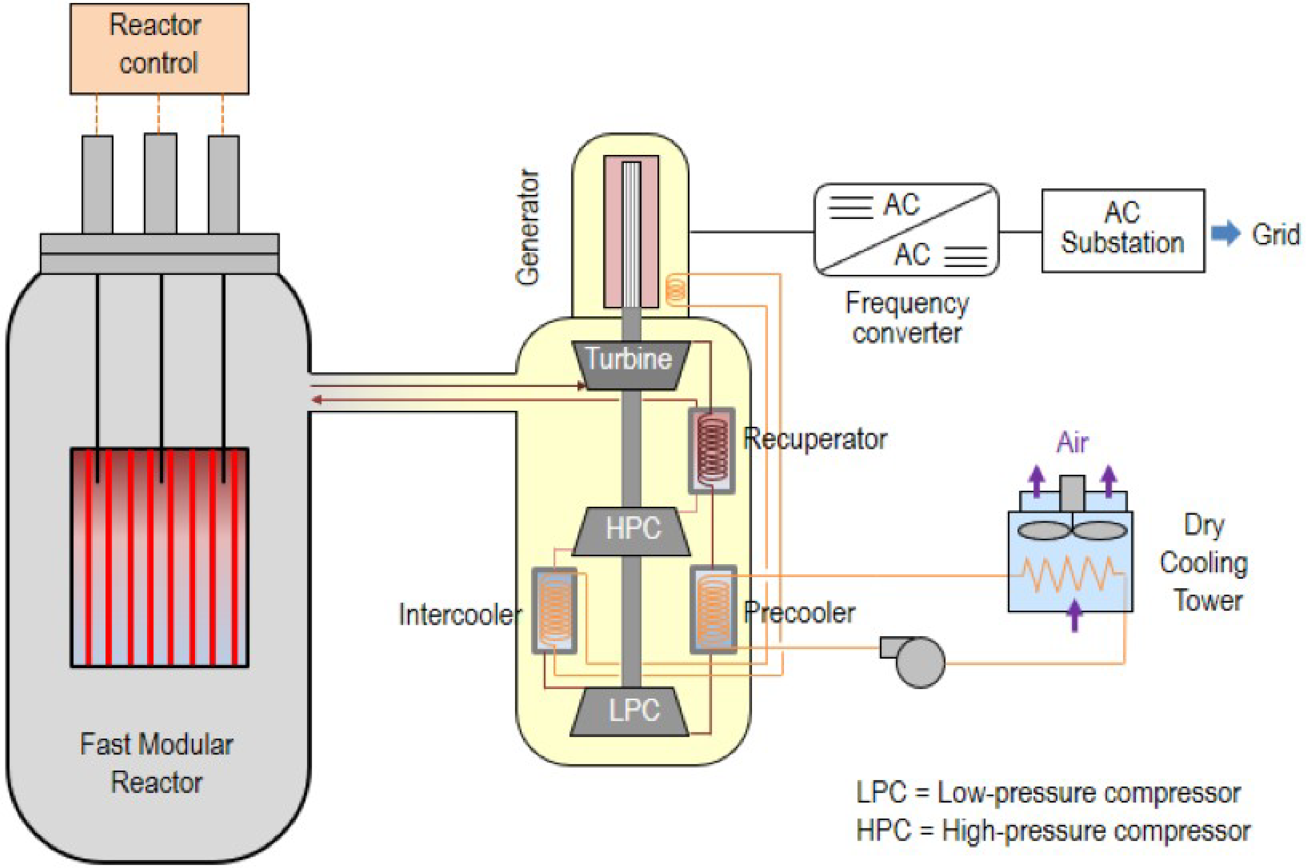
Fiber Braiding
Composite Fabrication
Dimensional Control
Pellet Loading
Joining/ Sealing

Vessel System Designed to Minimize Helium Leakage

- **Conceptual sizing calculation for normal and AOO conditions**
  - Design code is Section III, Division 5, 2021 Ed.
  - Thickness is adequate for operation up to 300,000 hours (~34 EFPY), will be extended to 540,000 hours (~60+ EFPY) (code revision)
  - Proven use of seal welds at joints to minimize helium leakage
- **Flow reductions during accident conditions reduces pressure loads**
- **Conceptual design complete on reactor vessel internals**

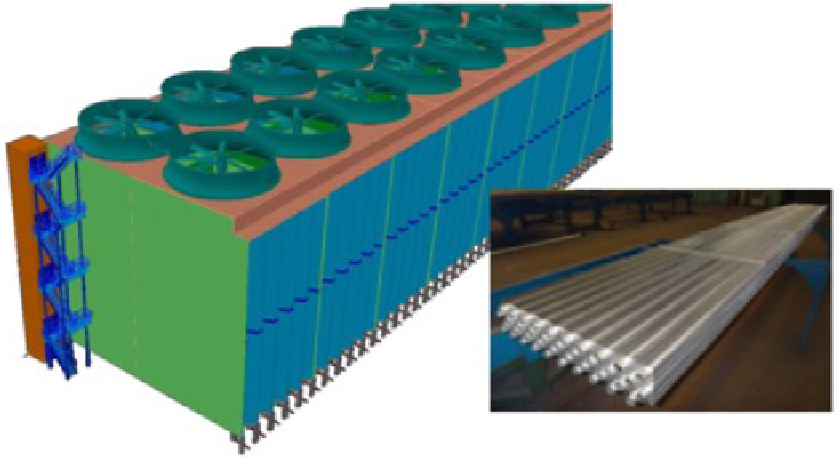


Power Conversion System (PCS) based on a Direct Brayton-Cycle



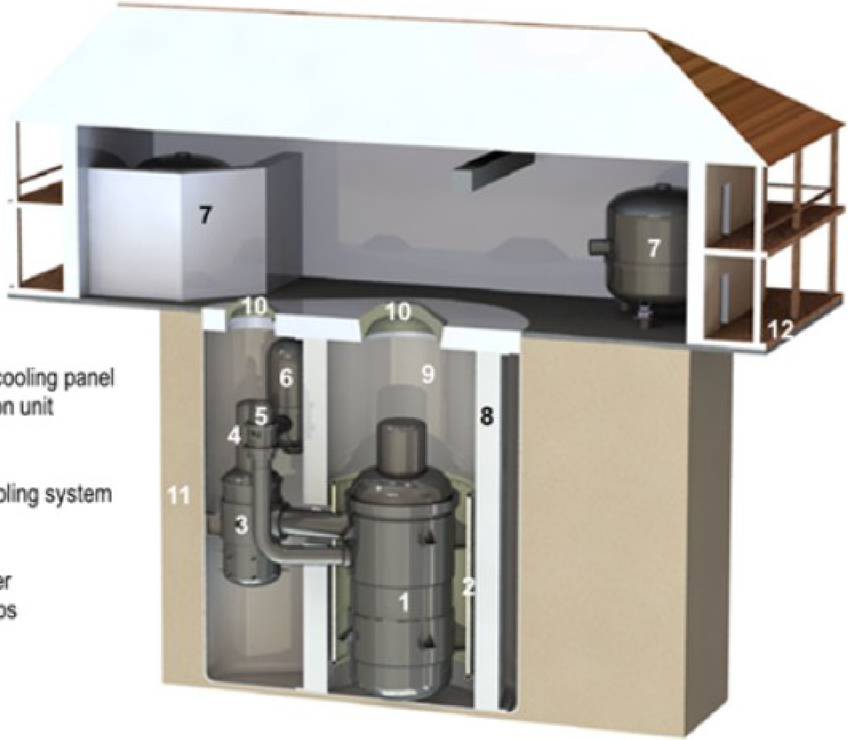
LPC = Low-pressure compressor
HPC = High-pressure compressor

- **Dry Cooling Tower**
  - Reduces impact on water resources and expands siting options



High-Efficiency Cycle that supports fast maneuvering capability

Containment Improves Safety and Siting



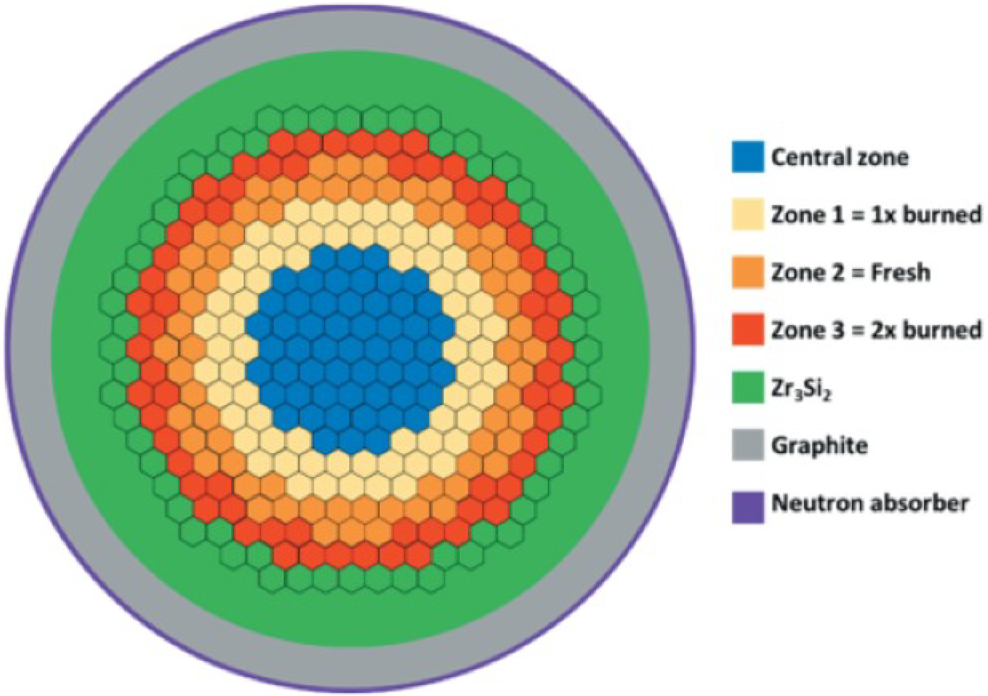
1. Reactor vessel
2. Reactor vessel cooling panel
3. Power conversion unit
4. Generator
5. Recirculator
6. Maintenance cooling system
7. Water tank
8. Containment
9. Containment liner
10. Maintenance caps
11. Concrete
12. Ground level

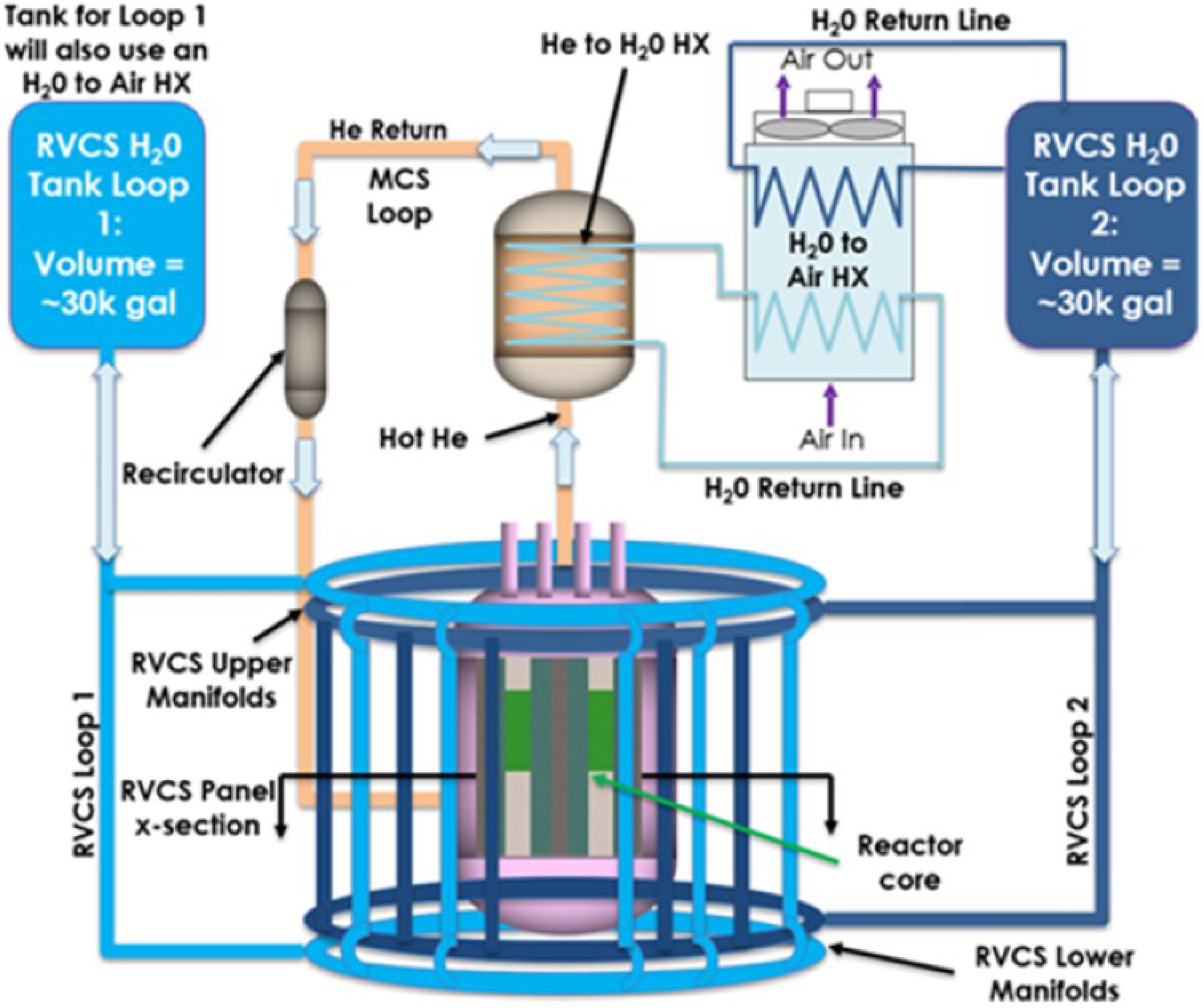
- **The Containment System (Category I Structure, SSE-qualified) includes below-grade, leak-tight Containment Vessel (multi-barrier, defense-in-depth)**
- **Need for containment heat removal, cleanup, and venting under investigation**

Below grade containment that is less vulnerable to airplane crashes

Residual Heat Removed By Active and Passive Systems

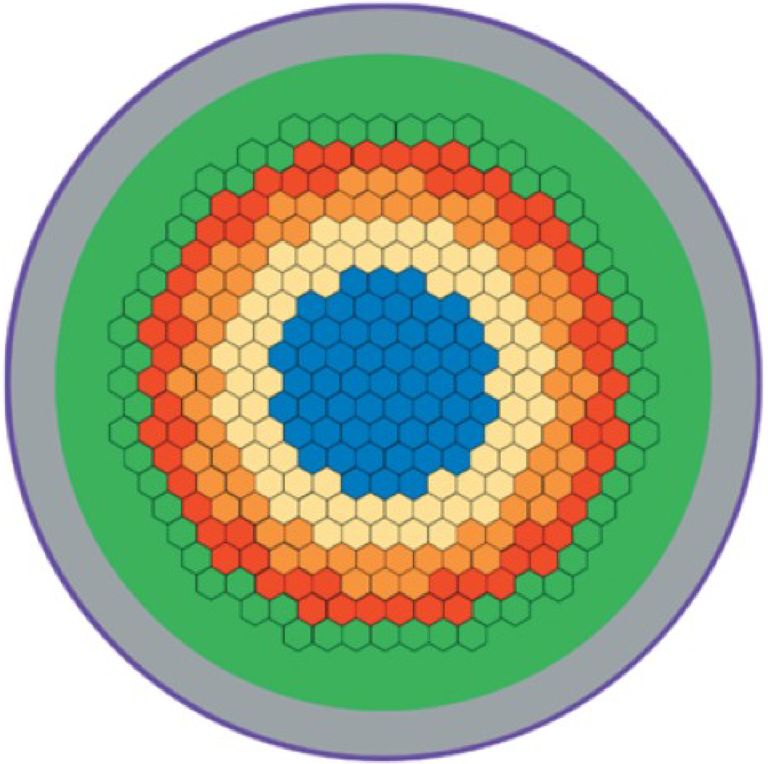
- Reactor Vessel Cooling System (2-loops)
- Maintenance Cooling System
- Annular core arrangement promotes passive heat removal

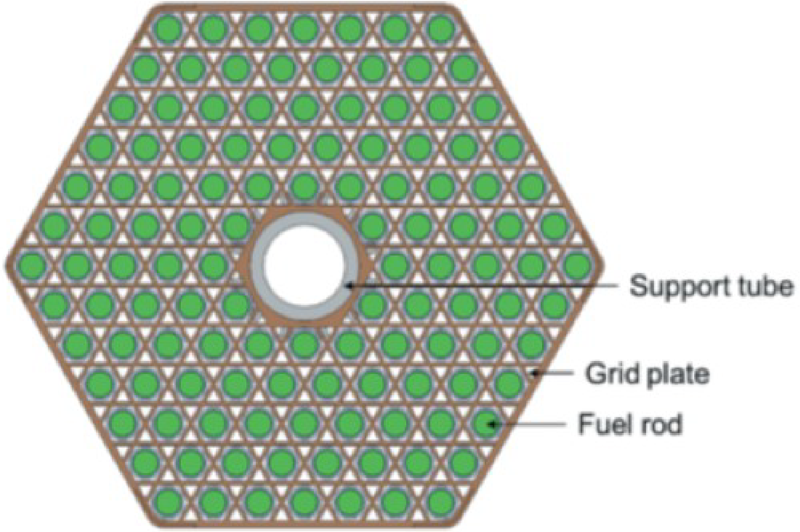


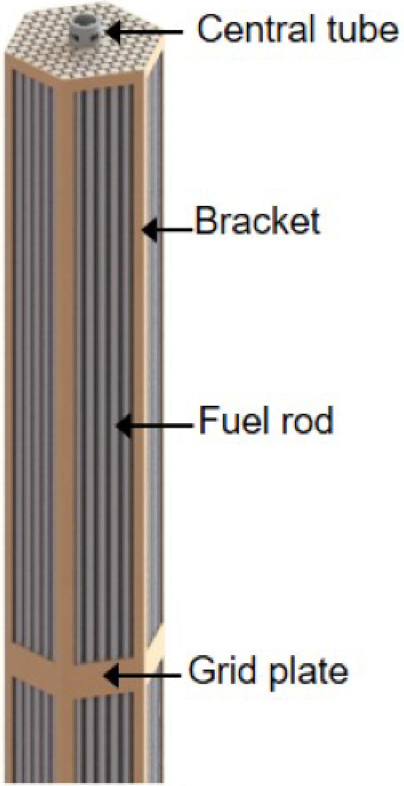


Core Bypass Flow Effects Normal Operation and Accidents

- **Core bypass flows through central and radial reflectors improves heat transfer during loss of forced cooling accidents**



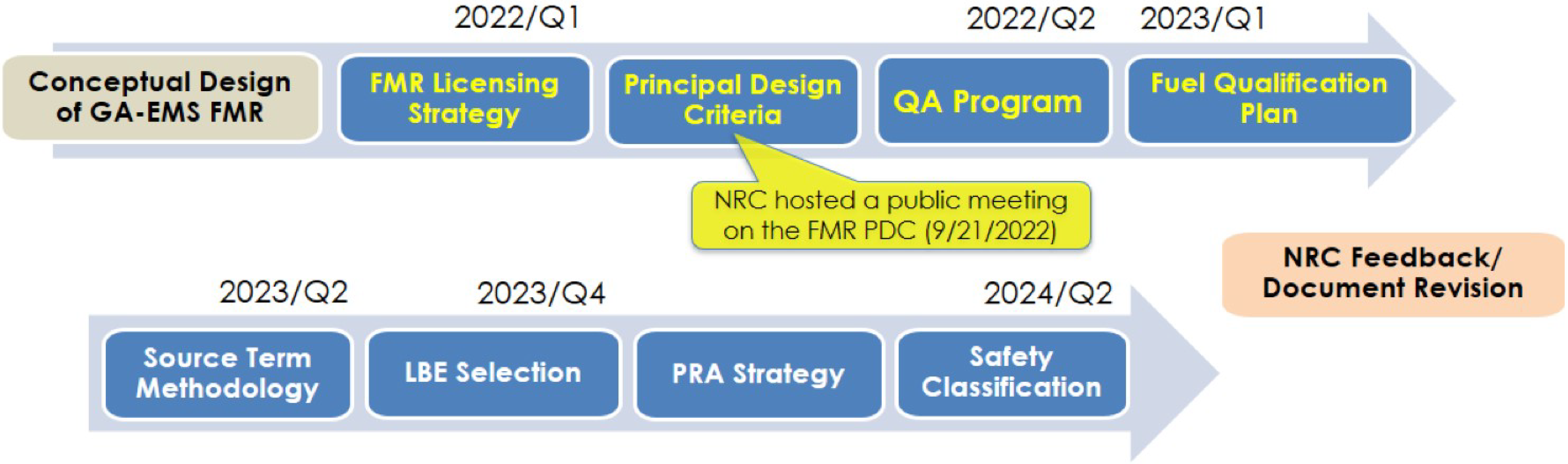




Progress in Pre-Application Licensing with NRC

GA-EMS has issued four documents and submitted them to NRC and DOE.

- Revision of the PDC document was issued following NRC's RAIs.
- The QA document was also revised following NRC's recommendations.



Conceptual Design of GA-EMS FMR

2022/Q1

FMR Licensing Strategy

Principal Design Criteria

NRC hosted a public meeting on the FMR PDC (9/21/2022)

2022/Q2

QA Program

2023/Q1

Fuel Qualification Plan

2023/Q2

Source Term Methodology

2023/Q4

LBE Selection

PRA Strategy

2024/Q2

Safety Classification

NRC Feedback/ Document Revision

Acknowledgements

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General Atomics Electromagnetic Systems

Fast Modular Reactor Principal Design Criteria

May 2, 2023

Presented To: Advisory Committee on Reactor Safeguards

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Principal Design Criteria Adapted From NRC Guidance

- **Regulatory Guide (RG) 1.232 established guidance for developing PDC for non-light-water reactors by modifying / supplementing 10 CFR 50, Appendix A, General Design Criteria (GDC) in three categories:**
  - Sodium-cooled fast reactors (SFR-DC)
  - Modular high-temperature gas-cooled reactors (MHTGR-DC)
  - Design-neutral advanced reactors (ARDC)
- **ARDC and MHTGR-DC used as starting point**
- **NRC rationale for adaptation of GDC modified for application to FMR-DC**

I. Overall Requirements – FMR DC 1 – 5

- **FMR-DC 1: Quality standards and records: Same as GDC**
- **FMR-DC 2: Design bases for protection against natural phenomena: Same as GDC**
- **FMR-DC 3: Fire protection: Same as ARDC**
- **FMR-DC 4: Environmental and dynamic effects design bases: Modified from MHTGR-DC**
  - "missiles" changed to "missiles originating both inside and outside the reactor helium pressure boundary" to cover turbomachinery
- **FMR-DC 5: Sharing of structures, systems, and components: Same as GDC**

II. Multiple Barriers – FMR-DC 10 – 19

- **FMR-DC 10: Reactor design: Fuel design using SiGA cladding functions like LWRs so SAFDL terminology used here and in other FMR-DCs**
- **FMR-DC 11, 13 - 15, 17 - 18: Same as ARDC and MHTGR-DC with minor terminology changes**
- **FMR-DC 12: Suppression of reactor power oscillations: The word “structures” added to address reflectors. The word “coolant” was deleted.**
- **FMR-DC 16: Containment design: Same as SFR-DC because FMR uses low-leakage, pressure-retaining containment**

III. Reactivity Control – FMR-DC 20 – 29

- **FMR-DC 20 - 24: Same as GDC**
- **FMR-DC 25: Protection system requirements for reactivity control malfunctions: Same as ARDC with minor terminology changes**
- **FMR-DC 26: Reactivity control systems: Combines GDC 26 and GDC 27 same as ARDC and MHTGR-DC**
- **FMR-DC 28: Reactivity limits: Same as MHTGR-DC**
- **FMR-DC 29: Protection against anticipated operational occurrences: Same as GDC**

IV. Fluid Systems – FMR-DC 30 – 46

- **FMR-DC 30 – 33: Same as MHTGR-DC**
- **FMR-DC 34: Residual heat removal: Similar to MHTGR-DC. Both active non-safety-related and passive safety-related systems available to remove residual heat. Incorporates requirements in GDC 35.**
- **FMR-DC 36 and 37: Same as MHTGR-DC**
- **FMR-DC 38 – 41: Same as ARDC**
- **FMR-DC 42: Inspection of containment atmosphere cleanup systems: Same as GDC**
- **FMR-DC 43, 45, 46: Same as ARDC**
- **FMR-DC 44: Same as MHTGR-DC**

V. Reactor Containment and VI. Fuel and Radioactivity Control

V. Reactor Containment – FMR-DC 50-57

- FMR-DC 50 – 53: Same as ARDC
- FMR-DC 54: Same as SFR-DC
- FMR-DC 55 - 57: Same as ARDC with minor terminology changes

VI. Fuel and Radioactivity Control – FMR-DC 60 – 64

- FMR-DC 60, 62, 63: Same as GDC
- FMR-DC 61: Same as ARDC

Acknowledgements

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General Atomics – Electromagnetic Systems Fast Modular Reactor Principal Design Criteria

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

Agenda

- General Atomics – Electromagnetic Systems (GA-EMS) Fast Modular Reactor (FMR) pre-application engagement
- GA-EMS FMR principal design criteria (PDC) topical report (TR) review timeline
- GA-EMS FMR design features
- PDC guidance
    - General Design Criteria (GDC)
    - Regulatory Guide (RG) 1.232
- GA-EMS PDC development approach
- Fast modular reactor design criteria (FMR-DC)
    - Impacts of key design choices on PDCs
    - FMR-DC overview
- Safety evaluation (SE) conclusions

GA-EMS FMR Pre-Application Engagement

Documents Submitted

Submittal	Document	Review Status
03/2022	Pre-Application Regulatory Engagement Plan	N/A
06/2022	PDC TR	Draft SE issued
02/2023	Fuel Qualification Plan Technical Report	Under review (white paper)
04/2023	Quality Assurance Program TR	Pending acceptance determination

Documents Expected

Submittal	Document
06/2023	Mechanistic Source Term Technical Report
12/2023	LBE Selection White Paper
05/2024	Safety Approach and Mini-PRA White Paper
05/2024	Safety Classification White Paper

*FMR demonstration expected by 2030 and deployment by mid-2030s

GA-EMS FMR PDC TR Review Timeline

- Submitted 06/06/22 (ML22154A555)
- Accepted 07/07/22 (ML22181B173)
- Requests for Additional Information (RAIs) issued 10/5/22 (ML22321A310)
- RAI response received 11/7/22 (ML22311A472)
- Revision 2 of TR transmitted 01/05/23 (ML23005A292)
- Draft SE issued 03/17/23 (ML23076A196)

GA-EMS FMR Design Features

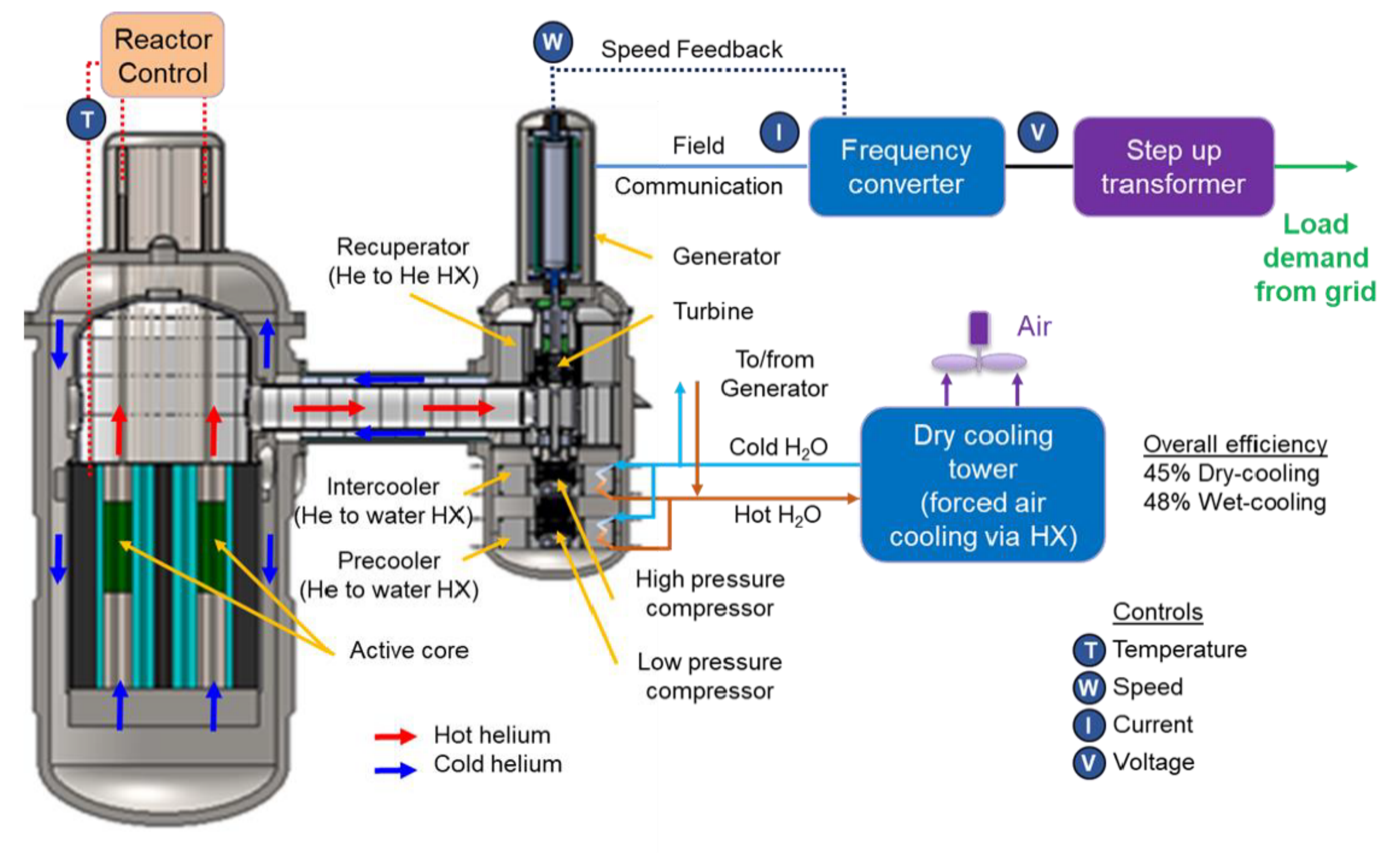

Figure 2. FMR Primary Coolant Flow and Power Generation

Source: REP, ML22087A510

GA-EMS FMR Design Features

1. Reactor vessel
2. Reactor vessel cooling panel
3. Power conversion unit
4. Generator
5. Recirculator
6. Maintenance cooling system
7. Water tank
8. Spent fuel storage
9. Containment
10. Containment liner
11. Maintenance caps
12. Concrete
13. Ground level

Figure 1. FMR Nuclear Island Components

Source: TR, ML22154A556

GA-EMS FMR Key Design Features

Feature	Design
Fuel	UO_2 pellets in silicon carbide fuel pins
Core arrangement	Pins in triangular pitch arranged into hexagonal bundles
Neutron spectrum	Fast
Coolant	Helium
Power conversion system	Gas turbine on primary coolant
Residual heat removal	Reactor vessel cooling system (water-fed, gravity-driven passive system)
Containment	Leak-tight containment building

PDC Guidance – 10 CFR 50 Appendix A GDC

“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**.”

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PDC Guidance – RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors”

- Issued April 2018 (ACRS letter March 2018)
- Documents three sets of acceptable PDCs:
  - Advanced reactor DC (ARDC) – generic, technology inclusive*
  - Sodium-cooled fast reactor DC (SFR-DC) – sodium-cooled fast reactors (e.g., PRISM)
  - Modular high temperature gas-cooled reactor DC (MHTGR-DC) – TRISO-fueled, helium-cooled, graphite-moderated HTGR

* For sodium/lead/gas-cooled fast reactors, modular high temperature gas reactors, fluoride high-temperature reactors, and molten salt reactors

GA-EMS Approach to PDC Development

- Start with ARDC, considering underlying safety basis
- If ARDC not fully applicable, assess SFR-DC and MHTGR-DC for direct adoption
- If SFR-DC or MHTGR-DC not directly applicable, apply DC that is most representative of FMR
- Adapt or refine selected DC

Key Design Feature Effects on PDCs

Feature	Design	Effect on PDCs
Fuel	UO_2 pellets in silicon carbide fuel pins	Use of specified acceptable fuel design limits (SAFDLs) instead of specified acceptable system radionuclide release design limits (SARRDLs)
Core arrangement	Pins in triangular pitch arranged into hexagonal bundles	
Neutron spectrum	Fast	Consider effect of structures on reactivity feedback
Coolant	Helium	Removal of coolant inventory control considerations consistent with MHTGR; use of reactor helium pressure boundary in lieu of reactor coolant pressure boundary
Power conversion system	Gas turbine on primary coolant	No particular effect
Residual heat removal	Reactor vessel cooling system (water-fed, gravity-driven passive system)	Adoption of MHTGR passive residual heat removal PDCs
Containment	Leak-tight containment building	Adoption of containment PDCs

FMR-DC – I. Overall Requirements

Criterion	Title	Basis PDC	Modified?
1	Quality standards and records.	ARDC	N
2	Design bases for protection against natural phenomena.	ARDC	N
3	Fire protection.	ARDC	N
4	Environmental and dynamic effects design bases.	MHTGR-DC	N
5	Sharing of structures, systems, and components	ARDC	N

FMR-DC – II. Multiple Barriers

Criterion	Title	Basis PDC	Modified?
10	Reactor design.	ARDC	Y - uses "heat removal" instead of "coolant"
11	Reactor inherent protection.	ARDC	N
12	Suppression of reactor power oscillations.	ARDC	Y - removes "coolant"
13	Instrumentation and control.	ARDC	Y - uses "helium pressure boundary" instead of "reactor coolant boundary"
14	Reactor helium pressure boundary.	MHTGR-DC	N
15	Reactor helium pressure boundary design.	MHTGR-DC	N
16	Containment design.	SFR-DC	N
17	Electric power systems.	MHTGR-DC	Y - uses SAFDLs instead of SARRDLs
18	Inspection and testing of electric power systems.	ARDC	N
19	Control room.	MHTGR-DC	N

FMR-DC – III. Reactivity Control

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

FMR-DC – IV. Fluid Systems (1)

Criterion	Title	Basis PDC	Modified?
30	Quality of reactor helium pressure boundary.	MHTGR-DC	N
31	Fracture prevention of reactor helium pressure boundary.	MHTGR-DC	N
32	Inspection of reactor helium pressure boundary	MHTGR-DC	N
33	[None - not applicable consistent with MHTGR-DC]	N/A	N/A
34	Residual heat removal.	MHTGR-DC	Y - includes both passive and active systems
35	[None - not applicable consistent with MHTGR-DC]	N/A	N/A
36	Inspection of passive residual heat removal system.	MHTGR-DC	N
37	Testing of residual heat removal system.	MHTGR-DC	Y - includes both passive and active systems
38	Containment heat removal.	ARDC	N
39	Inspection of containment heat removal system.	ARDC	N

FMR-DC – IV. Fluid Systems (2)

Criterion	Title	Basis PDC	Modified?
40	Testing of containment heat removal system.	ARDC	N
41	Containment atmosphere cleanup.	ARDC	N
42	Inspection of containment atmosphere cleanup systems.	ARDC	N
43	Testing of containment atmosphere cleanup systems.	ARDC	N
44	Structural and equipment cooling.	ARDC	N
45	Inspection of structural and equipment cooling systems.	ARDC	N
46	Testing of structural and equipment cooling systems.	ARDC	N

FMR-DC – V. Reactor Containment

Criterion	Title	Basis PDC	Modified?
50	Containment design basis.	ARDC	N
51	Fracture prevention of containment pressure boundary.	ARDC	N
52	Capability for containment leakage rate testing.	ARDC	N
53	Provisions for containment testing and inspection.	ARDC	N
54	Piping systems penetrating containment.	SFR-DC	Y - removes "reactor"
55	Reactor helium pressure boundary penetrating containment.	ARDC	Y - uses "helium pressure boundary" instead of "reactor coolant boundary"
56	Containment isolation.	ARDC	N
57	Closed system isolation valves.	ARDC	Y - uses "helium pressure boundary" instead of "reactor coolant boundary"

FMR-DC – VI. Fuel and Reactivity Control

Criterion	Title	Basis PDC	Modified?
60	Control of releases of radioactive materials to the environment.	ARDC	N
61	Fuel storage and handling and radioactivity control.	ARDC	N
62	Prevention of criticality in fuel storage and handling.	ARDC	N
63	Monitoring fuel and waste storage.	ARDC	N
64	Monitoring radioactivity releases.	ARDC	N

Safety Evaluation Conclusions

- GA-EMS appropriately considered RG 1.232 and developed a sufficient set of PDCs appropriate for establishing requirements for the FMR design.
- PDCs establish the necessary design, fabrication, construction, testing, and performance design criteria for safety-significant SSCs to provide reasonable assurance that an FMR could be operated without undue risk to the health and safety of the public. (10 CFR 50 App A)
- This TR can be used by future FMR applicants, but if the reactor design differs from that discussed in the TR use of the PDCs in the TR must be justified.

FMR-DC Summary

- Directly adopted from RG 1.232
  - From ARDC: FMR-DC 1, 2, 3, 5, 11, 18, 20, 21, 22, 23, 24, 25, 29, 38, 39, 40, 41, 42, 43, 44, 45, 46, 50, 51, 52, 53, 60, 61, 62, 63, 64
  - From SFR-DC: FMR-DC 16
  - From MHTGR-DC: FMR-DC 4, 14, 15, 19, 28, 30, 31, 32, 36
- Modified from RG 1.232
  - FMR-DC 10 (ARDC 10), 12 (ARDC 12), 13 (ARDC 13), 17 (MHTGR-DC 17), 26 (ARDC 26), 34 (MHTGR-DC 34), 37 (MHTGR-DC 37), 54 (SFR-DC 54), 55 (ARDC 55), 57 (ARDC 57)

FMR-DC Modified from RG 1.232

ARDC 10	FMR-DC 10
Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	*Reactor design.* The reactor core and associated ~~coolant~~ heat removal, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Basis: Helium inventory control is not necessary to meet SAFDLs due to reactor system design; consistent with MHTGR-DC (which use SARRDLs instead) and other FMR-DC

FMR-DC Modified from RG 1.232

ARDC 12	**FMR-DC 12**
Suppression of reactor power oscillations. The reactor core; associated structures; and associated coolant, control, and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.	*Suppression of reactor power oscillations.* The reactor core~~;~~, associated structures~~;~~, and associated ~~coolant, ~~control~~,~~ and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Basis: Helium coolant does not have a significant effect on reactivity for the FMR

FMR-DC Modified from RG 1.232

ARDC 13	FMR-DC 13
Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	*Instrumentation and control.* Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety, including those variables and systems that can affect the fission process~~,~~ and the integrity of the reactor core, the reactor ~~coolant~~ helium pressure boundary, and the containment and ~~its~~ associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Basis: More appropriate to say “reactor helium pressure boundary” than “reactor coolant boundary” for FMR, consistent with MHTGR-DC and other FMR-DC

FMR-DC Modified from RG 1.232

ARDC 26	FMR-DC 26
Reactivity control systems. A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the design limits for the fission product barriers are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the design limits for the fission product barriers are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.	*Reactivity control systems.* A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the design limits for the fission product barriers are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that the design limits for the fission product barriers are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.

Basis: GDC 26 includes explicit consideration of Xe burnout; while Xe is not expected to be a significant reactivity contributor in the FMR it is not incorrect to explicitly include it

FMR-DC Modified from RG 1.232

MHTGR-DC 34	**FMR-DC 34**
Passive residual heat removal.	~~Passive r~~Residual heat removal.
A passive system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable system radionuclide release design limits and the design conditions of the reactor helium pressure boundary are not exceeded.	~~A passive s~~System(s) to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable ~~system radionuclide release~~ fuel design limits and the design conditions of the reactor helium pressure boundary are not exceeded.
During postulated accidents, the system safety function shall provide effective cooling.	During postulated accidents, the system safety function shall provide effective core cooling.
Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure the system safety function can be accomplished, assuming a single failure.	Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure the system safety function can be accomplished, assuming a single failure.

Basis: The MHTGR included a passive residual heat removal (RHR) system because of the low core power density. FMR has multiple RHR systems including active non-safety-related systems and passive safety-related systems, and the DC should be broad enough to apply to all of them.

FMR-DC Modified from RG 1.232

MHTGR-DC 37	FMR-DC 37
Testing of passive residual heat removal system. The passive residual heat removal system shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including associated systems, for AOO or postulated accident decay heat removal to the ultimate heat sink and, if applicable, any system(s) necessary to transition from active normal operation to passive mode.	*Testing of ~~passive~~ residual heat removal system.* The ~~passive~~ residual heat removal system(s) shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including associated systems, for AOO or postulated accident decay heat removal to the ultimate heat sink and, if applicable, any system(s) necessary to transition from active normal operation to passive mode.

Basis: The MHTGR included a passive residual heat removal (RHR) system because of the low core power density. FMR has multiple RHR systems including active non-safety-related systems and passive safety-related systems, and the DC should be broad enough to apply to all of them (same as FMR-DC 34).

FMR-DC Modified from RG 1.232

SFR-DC 54	FMR-DC 54
Piping systems penetrating containment. Piping systems penetrating the reactor containment structure shall be provided with leak detection, isolation, and containment capabilities that have redundancy, reliability, and performance capabilities necessary to perform the containment safety function and that reflect the importance to safety of preventing radioactivity releases from containment through these piping systems. Such piping systems shall be designed with the capability to verify, by testing, the operational readiness of any isolation valves and associated apparatus periodically and to confirm that valve leakage is within acceptable limits.	*Piping systems penetrating containment.* Piping systems penetrating the ~~reactor~~ containment structure shall be provided with leak detection, isolation, and containment capabilities that have redundancy, reliability, and performance capabilities necessary to perform the containment safety function and that reflect the importance to safety of preventing radioactivity releases from containment through these piping systems. Such piping systems shall be designed with the capability to verify, by testing, the operational readiness of any isolation valves and associated apparatus periodically and to confirm that valve leakage is within acceptable limits.

Basis: There are other major SSCs other than just the reactor within containment (e.g., the power conversion system) so it is appropriate to remove the word "reactor"

FMR-DC Modified from RG 1.232

ARDC 55	FMR-DC 55
Reactor coolant boundary penetrating containment. Each line that is part of the reactor coolant boundary and that penetrates the containment structure shall be provided with containment isolation valves, as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: …	*Reactor ~~coolant~~ helium pressure boundary penetrating containment.* Each line that is part of the reactor ~~coolant~~ helium pressure boundary and that penetrates the reactor containment structure shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: …

Basis: More appropriate to say “reactor helium pressure boundary” than “reactor coolant boundary” for FMR, consistent with MHTGR-DC and other FMR-DC

FMR-DC Modified from RG 1.232

ARDC 57	FMR-DC 57
Closed system isolation valves. Each line that penetrates the containment structure and is neither part of the reactor coolant boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve, unless it can be demonstrated that the containment safety function can be met without an isolation valve and assuming failure of a single active component. The isolation valve, if required, shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.	*Closed system isolation valves.* Each line that penetrates the containment structure and is neither part of the reactor ~~coolant~~ helium pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve unless it can be demonstrated that the containment safety function can be met without an isolation valve and assuming failure of a single active component. The isolation valve, if required, shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Basis: More appropriate to say “reactor helium pressure boundary” than “reactor coolant boundary” for FMR, consistent with MHTGR-DC and other FMR-DC