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(202) 234-4433

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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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706TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

OPEN SESSION

+ + + + +

WEDNESDAY

JUNE 7, 2023

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The Advisory Committee met via hybrid In-Person and Video-Teleconference, at 1:00 p.m. EDT, Joy L. Rempe, Chairman, presiding.

COMMITTEE MEMBERS:

JOY L. REMPE, Chairman

WALTER L. KIRCHNER, Vice Chairman

DAVID A. PETTI, Member-at-Large

RONALD G. BALLINGER, Member

CHARLES H. BROWN, JR., Member

VICKI M. BIER, Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

JOSE MARCH-LEUBA, Member

MATTHEW W. SUNSERI, Member

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

ALSO PRESENT:

REED ANZALONE, NRR

JOHN BOLIN, GA-EMS

SAMUEL CUADRADO DE JESUS, NRR

STEVE JONES, NRR

SCOTT MOORE, Executive Director, ACRS

ANDREW PROFFITT, NRR

A G E N D A

Opening Remarks by the ACRS Chairman

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

1:05 p.m.

CHAIR REMPE: So it is past 1:00 p.m. And I apologize, but there was a technical difficulty in the room. And the ACRS meeting will now come to order. This is the first day of the 706th meeting of the Advisory Committee on Reactor Safeguards.

And I am Joy Rempe, Chairman of the ACRS. Other members are Ron Ballinger, Vicki Bier, Charles Brown, Vesna Dimitrijevic, Greg Halnon, Walt Kirchner, Jose March-Leuba, Dave Petti, and Matt Sunseri. We do have a quorum. And today the committee is meeting in person and virtually.

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

The committee provides its advice on safety matters to the Commission through its publicly available letter reports. The Federal Register notice announcing this meeting was published on May 12th, 2023. This announcement provided a meeting agenda as well as instructions for interested parties to submit written documents or request opportunity to address the committee. The DFO for today's meeting is Weidong Wang.

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

We received no written comments or requests to make oral statements from members of the public regarding today's session. Periodically the meeting will be opened to accept comments from participants listening to our meeting. Written comments may be forwarded to Mr. Weidong Wang, today's DFO.

During today's meeting, the committee will consider the General Atomics Fast Modular Sensible Design Criteria topical report. Note that portions of the GA session may be closed as stated in the agenda.

A transcript of the open portions of the meeting is being kept, and it's requested that speakers identify themselves and speak with sufficient clarity and volume so they may be readily heard. Participants should mute themselves when they're not speaking.

Before we begin today's session, I do have two topics of interest. First, we'd like to welcome Paris Bradley, a summer hire, who's pursuing a Master's Degree in Nuclear Science and Engineering at the Colorado School of Mines.

And second, I'd like to ask members to join me in congratulating Member Petti for his reappointment to ACRS for a second term.

I'd also like to thank Scott and Alisha's organization for their efforts to get this package together so that we could have this renomination occur.

And at this time, I'd like to ask some of the other members if they have any opening remarks. Hearing none, I'd like to ask --

MR. MOORE: Chairman?

CHAIR REMPE: Oh, they do --

MR. MOORE: Yeah. I'm not a member but Executive Director, Scott Moore. I'd also like to let the members know, and recognize we have a new member and, excuse me, we have a new staff member in PMDA. Tyesha Bush joined us about three weeks ago. And she's working with a future incoming member right now. But, yeah.

CHAIR REMPE: Thank you, Scott. I did meet Tyesha today, and we're glad to have her onboard. She's already been very helpful to us.

MR. MOORE: Thanks.

CHAIR REMPE: So at time then, not hearing anyone else with an opening remark, I'd like to ask Member Bier to lead us through our first topic. Well, I think I'll ask Member Bier to start, and then she will call on them. Okay. Thank you.

MEMBER BIER: All right. I'm Vicki Bier, I'm Chair of the General Atomics Subcommittee for ACRS. And we had an overview of these issues in May of this year, last month. And I'm pleased to be hearing from both the NRC staff and GA again here full committee this week.

And Andrew Proffitt from Nuclear Reactor Regulations will be giving the NRC's introductory remarks.

MR. PROFFITT: Yeah, thank you, Member Bier. This is Andrew Proffitt from the NRC staff, acting chief of the Advanced Reactor Licensing Branch, and I'll give a brief overview of the presentation today.

So we're currently engaged with General Atomics Electromagnetic Systems in pre-application activities related to an expected application for their 50 megawatt electric fast modular reactor design.

Currently we have two topical reports and one white paper under review. The PDC topical report we're here to talk about today, we're also reviewing a Quality Assurance Program topical report and a fuel qualification plan white paper.

We're expecting several more over the next year related to white papers related to mechanistic source term, licensing basis event selection, safety approach and PRA, and safety classification.

So GA has undertaken these activities in pursuit of a FMR demonstration by 2030 and deployment in the mid-2030s. The staff's looking forward to continued interactions with the committee on these topics as we move forward on this application, and other advanced reactor applications, and appreciate the opportunity to be here. Thank you, Member Bier.

MEMBER BIER: Okay. Yes?

VICE CHAIR KIRCHNER: Andrew is it your

anticipation that these white papers will eventually evolve into TRs to support their subsequent application?

MR. PROFFITT: Many times they do, and specifically on these topics. These are some of the more complicated topics we deal with in pre-application. And we certainly encourage them to turn into topical reports. I mean, that's not a requirement, but we do have, right now, a draft white paper out on pre-application engagement that outlines a lot of the topics we'd like to see in the topical report space.

And we commit that if applicants or potential applicants meet those, what we lay out in that draft white paper, that we'll accelerate their review when they do come into play.

VICE CHAIR KIRCHNER: Thank you.

MEMBER BIER: Any other questions or comments at this stage?

Okay. If not, I am happy to turn this over to John Bolin, a senior staff engineer at GA, who will be giving the GA presentation virtually. So you can go ahead whenever you're ready, John.

MR. BOLIN: Okay, thank you. So I'm going to give everyone an overview of the Fast Modular

Reactor Conceptual Design. We are midway through a conceptual design effort, a cooperative agreement with the Department of Energy.

Let's see, I'm trying to see how to advance my slides. Okay. This slide covers some of the major parameters of the fast modular reactor and compares those with the gas turbine modular heating reactor and the Westinghouse AP1000.

The main thing to note is the small thermal output, 100 megawatts thermal. We have a relatively small core power density, and in particular we have a fuel rod linear power that is about eight times lower than the AP1000. And we also have a relatively flat axial and radial power distribution that limits our hot channel power factor to 1.52.

The design, and we'll go over this in a little more detail, the design uses a high density UO2 fuel in a silicon carbide composite cladding. And we'll go over that in subsequent slides.

So like I mentioned, the fuel is one of the key components, one of the first barriers to fission product release. And the fuel leverages UO2 legacy development and also SiGA, sodium carbide composite cladding development.

So we are using high density UO2 that's

been proven in light water reactors and also tested in fast reactors. The silicon carbide composite cladding that we will be using in our fuel is undergoing testing and maturation through the DOE Accident Tolerant Fuel Program, and that includes current testing that's going on in ATR.

The fuel design uses the ATF-LWR dimensions, but unlike a light water reactor fuel rod, we have a large plenum. Approximately one-third of the fuel link is a plenum, similar to the legacy liquid metal fast reactor fuel design. And so here we see images of the fuel assembly, it's a hexagonal fuel assembly, and then pictures of the silicon carbide composite cladding.

We have manufactured test rodlets that will be inserted in ATR. And I'll go over that in a little more detail in the next slide. And we have an X-ray image of the cladding tube. And then the endcap is sealed at the end of the cladding tube.

This goes into a little more detail on the fuel, since the fuel is a critical component and, like I said, one of the first barriers to fission product release. We are working with both Argonne and Idaho in developing the fuel and in also testing the fuel.

We have manufactured both standard and

compact rodlets. And compact rodlets have a higher enrichment but a reduced size compared to the standard rodlets. And so with the compact rodlets we can do accelerated irradiation in ATR and get to a full burn up.

ANL is working with us on a BISON model and is analyzing both the standard reactor and also the rodlets. And of course INL is assembling, will be assembling our rodlets into a capsule that will be inserted into ATR and go through between three to six cycles of ATR radiation. And afterwards, we'll have a post-radiation examination.

This image goes through some of the procedures associated with fuel fabrication. And we are using established ATF fabrication procedures. The left side is pretty much standard UO2, high density UO2 fabrication. And then on the right side is the fabrication steps for sodium carbide composite cladding, including the final joining and sealing of the endcap.

Should I pause for questions, or should I save questions to the end?

CHAIR REMPE: I think we'll continue to interrupt you as we have in other times, but since you have paused, I have a question.

Maybe it's not crossed your radar, but your use of the DOE codes, is that going to -- have you thought forward if, at some point when you have this reactor and you're going to be trying to market around the world, have you thought about will that cause any complications with transmitting those codes for the future owner operators of the plant to use it? Or is that something that hasn't crossed your radar yet?

MR. BOLIN: That has not crossed our radar yet. In particular, I think you're referring to BISON, of course?

CHAIR REMPE: Well, yeah. And we've had other applicants come in and, at some point, some of these codes are not going to be, you know, how will one get them to something where you can transmit it and sell it to the owners and operators? And, you know, is it exportable? Those kinds of questions that I assume it's too early to be thinking about that.

MR. BOLIN: We haven't thought about that. And whether BISON would be part of that package, I don't know.

Okay. So now I want to go through some of the other major components of the design, particularly components that are part of the defense in depth. So the vessel system is part of, of course, the primary coolant pressure boundary. And we have done conceptual sizing for both normal operations and AOO conditions.

We are basing it on the ASME Code, Section III, Division 5. And the thickness is adequate for operation up to 300,000 hours. And we plan on extending that with the subsequent code revision to 540,000 hours, almost 60 years of effective full power operation. And we'll be using COL at joints to minimize heating leakage, because heating leakage is an economic penalty for us.

One interesting thing to note is when we do have accident conditions, or even reduced load conditions, that actually reduces the pressure load on the vessel system, so that's an added benefit. And so far, we are complete on the conceptual design of the reactor vessel internals.

This image also shows the general flow path through the reactor. We do have a cross vessel connecting the reactor vessel to the power conversion unit. So hot helium exits through the top of the core, through that cross vessel, returns through the outer portion of that, and goes down to the bottom of the core, and then flows up through the fuel assemblies.

I should also note that we do have a zirconium silicide reflector adjacent to the core that minimizes, of course, neutron leakage, but it also minimizes the reduction in neutron spectrum so that we remain a fast spectrum reactor, and without much thermalization of the neutrons that might physically be found in other kinds of reflectors.

An important part of our design is the power conversion system. And this power conversion system is a direct Brayton cycle, inter-cooled. So we see here in this image more on the flow path through the reactor, but also through the power conversion system.

Because it's inter-cooled, the hot helium first goes to the turbine, then goes to a recuperator, pre-cooler, a low pressure compressor, an inter-cooler, and then a high pressure compressor, then back through the recuperator, and then back to the reactor.

So we have a permanent magnet motor generator that allows us to operate our reactor asynchronously so we can vary the speed and flow rate through the reactor and quickly adjust the power and flow rate to match the grid demand. So that allows us to be able to provide grid stability and load following services to the grid.

We also are using a dry cooling tower as our standard ultimate heat sink for our power conversion system. And that, of course, reduces impact on water resources and expands our citing options.

And then the final barrier to fission product release is the containment. And unlike standard modular heating reactors, we actually have a leak-tight containment vessel. It is below grade, similar to all other modular heating reactors. So below grade obviously makes us less vulnerable to airplane crashes, and it's leak-tight so that we can tolerate some fuel failure during extreme accidents and still meet strict offsite dose limits.

We're still investigating whether we need containment heat removal, containment fission product cleanup, and venting post-accident. Those things are probably maybe needed for expediting post-accident recovery, but we are still looking to see whether that's actually required to meet dose limits. So they may not be safety related at all.

VICE CHAIR KIRCHNER: John, this is Walt Kirchner.

MR. BOLIN: Yes.

VICE CHAIR KIRCHNER: Can you just give us an idea of your containment design pressure?

MR. BOLIN: We are looking at a design pressure of normally about 0.6 megapascals but with an upper limit of 0.7 megapascals, or seven atmospheres.

So with the intention of being within standard light water reactor containment design capabilities, that was or purpose in setting those pressures.

VICE CHAIR KIRCHNER: Yeah, thank you. So your helium inventory is such that you don't get a much higher pressure --

MR. BOLIN: Correct, correct.

VICE CHAIR KIRCHNER: -- for the major rupture of the --

(Simultaneous speaking.)

MR. BOLIN: So with a depressurization event, we will be within those design pressures, correct.

So one of the key safety systems is, of course, residual heat removal, safety functions. And we are doing that by both active and passive. So was there another question?

Okay. So we're doing that by both active and passive systems. The main passive system is reactor vessel cooling systems, so that would be a safety related system. But it has both safety related and non-safety related components to it.

The RVCS water tanks provide a seven-day supply of cooling, even if only one of them is available, provide a seven-day supply. Those tanks are kept cool by a forced circulation system and a water tower, or water cooling tower.

The RVCS loop, there's two of them. So there is a redundancy in those loops. They operate passively by natural circulation, a buoyancy driven flow to the water tank. There is also a maintenance cooling system.

It's an active system that is primarily there for maintenance cooling but is also available if the power conversion system fails and is not able to provide force convection cooling. But that maintenance cooling system is a not safety related system. So it has typically, I mean, it's sort of a typical configuration of a helium to water heat exchanger, and every circulator that circulates hot helium through the heat exchanger and then back to the core.

We also have arranged the core in an annular configuration, and that promotes also passive heat removal to the vessel and then from the vessel to the reactor vessel cooling system panels.

MEMBER HALNON: John, can you point to what would be safety related in this RVCS?

MR. BOLIN: Let's see?

MEMBER HALNON: Well, for example, the water towers, are they safety or non-safety?

MR. BOLIN: The water towers would not be safety related --

MEMBER HALNON: Yeah, because that's kind of a keep cool system type of thing.

MR. BOLIN: That's to keep it cool. And obviously it's electric powered. And also the pump that pumps the water from the RVCS -- so in the RVCS tank there is a heat exchanger, a water to water heat exchanger that has a pump that pumps water through that heat exchanger, and out to the towers, and back. So all those electrical systems are not safety related.

The tank is safety related, and obviously the pipes feeding RVCS --

MEMBER HALNON: The tanks will have some temperature limit on it based for pre-existing conditions to keep it. And that's why you have the keep cool system.

MR. BOLIN: Well, and the passive heat removal relies on boil off of that water to remove heat from the RVCS system.

MEMBER HALNON: Okay. So I think I get it. Thanks.

VICE CHAIR KIRCHNER: Just a follow on to Greg's question, on the recirculator, is there an isolation valve or some --

(Simultaneous speaking.)

VICE CHAIR KIRCHNER: -- where do you draw the line on the safety related part of the primary coolant pressure boundary?

MR. BOLIN: The pressure boundary of the maintenance cooling system would be safety related. But the function of the maintenance cooling system, the heat exchanger and the circulator themselves, are not safety related. So there is no isolation valves in the maintenance cooling system. There will be a flow shutoff valve so that we don't have flow through the maintenance cooling system during normal operation. But otherwise, I think that answers your question, doesn't it?

VICE CHAIR KIRCHNER: Yes, it does. Yeah, I just was checking on the pressure boundary.

MR. BOLIN: Yes. So the maintenance

cooling system is located in the containment. And the pressure boundary itself would be safety related.

And that concludes my presentation. I just want to acknowledge that this, like I said, this is supported by the U.S. Department of Energy under their Advanced Reactor Concepts 2020 program.

CHAIR REMPE: John?

MR. BOLIN: Yes?

CHAIR REMPE: I know that the staff's going to talk about what you've selected for your principal design criteria. During the subcommittee meeting, I mentioned about the critical safety functions.

But again, as we go through a lot of these new applications coming in, I think it's good for us to understand how you came up with the principal design criteria and if you started with looking at what the critical safety functions were, and then kind of linking them to the principal design criteria, to make sure that you'd identified enough principal design criteria.

And could you talk a little bit about the process that you followed?

MR. BOLIN: So we did leverage our past work on both GT-MHR and the energy multiplier module.

We had looked at the energy multiplier module. It has a lot of similarities to the FMR design.

And the critical safety functions there are where we started. The functions of controlling heat generation, removing heat, and preventing chemical attack on the fuel. And so those are the critical functions that we started with.

And in developing the principal design criteria, and in developing the design itself, you know, we made every effort to incorporate defense-in-depth into our design. That's why, in particular, we do have a leak-tight containment as our ultimate barrier to fission product release.

CHAIR REMPE: Thank you, that helps.

MEMBER BIER: Any other questions or comments for John before we hear from the staff on the principal design criteria?

MEMBER PETTI: I just had one, John. Did you map your criteria to the safety functions? I know there's a lot of criteria. It would be interesting to know how many of them are related to heat removal, how many are related to controlled chemical attack --

MR. BOLIN: We did not explicitly map the PDCs to the safety functions. I mean, that's an exercise we could do, but we did not do that during our development process.

MEMBER PETTI: Okay.

MEMBER BIER: Okay. With that then, I think we are ready to hear from staff. And, Reed, you will present the staff's NCR?

MR. ANZALONE: That's right. Let me get a second to set up here.

MEMBER BIER: Oh, absolutely. Take your time. Thank you.

MR. ANZALONE: One of my key lessons from last time is that I need to get a lot closer to the microphone.

MEMBER BIER: Yes, absolutely. And I think we're still amazingly ahead of schedule, so we're good.

MR. ANZALONE: Okay. So first of all, I want to thank the committee for having me here today and for your time on this important topic. My name is Reed Anazalone. I'm a senior nuclear engineer in the Office of Nuclear Reactor Regulation, Division of Advanced Reactors and Non-Power Production and Utilization Facilities, Advanced Reactor Technical Branch 2.

My colleagues who helped on this report were Sam Cuadrado, sitting on the side over there who was the project manager, Sheila Ray, who handled the electrical related PDCs, and Steve Jones who handled the containment PDCs.

I wanted to kind of reflect after John's presentation on how this was the very first thing that General Atomics submitted to us after the regulatory engagement plan. And so, you know, we think that this was an appropriately early engagement on PDCs.

And we hope that the PDCs will really drive the design in the direction that considers, you know, safety as one of the critical aspects and really will help to establish the design of the facility going forward. So we think they got enough of the design done to be able to establish what the PDCs ought to be and go on from there.

I have to figure out how to advance the slides.

So for my presentation today, I'll be talking a little bit about the requirements and guidance that exists for principal design criteria. I'll briefly touch on the development approach that General Atomics presented to us in the topical report.

And then I'll go into the fast modular reactor design criteria themselves. And I'll talk a little bit about what the design choices we perceived

General Atomics made and how those affected the principal design criteria. And then I'll go through a brief overview of the design criteria themselves. Then I'll talk about the safety evaluation conclusions briefly.

So the guidance for -- first of all, talk about the requirements for PDCs, which isn't on this slide. And I believe that General Atomics is pursuing a Part 50 pathway for this initial license. And so they have a requirement to submit principal design criteria under 5034.

And then Appendix A establishes, in this first excerpt, which I won't read, what the PDCs are required to do, what the scope has to be. It has to establish the design fabrication, construction testing, and performance requirements for SSCs that are important to safety.

And then it also, for non-light water reactors, or reactors that are different from the water-cooled nuclear power plants similar in design and location to plants which construction permits have been issued, provides guidance in establishing what the PDCs ought to look like.

But the staff also issued more specific guidance in developing principal design criteria for non-light water reactors. This was issued in April of 2018. I think this is one of the earlier applications that we're actually seeing of this reg guide.

The reg guide documents three acceptable sets of principal design criteria. There is a generic, technology inclusive set of PDCs called the Advance Reactor Design Criteria, or ARDCs, and it's technology inclusive.

And I have a caveat there that it's technology inclusive for the certain types of technologies that it was designed to be inclusive for, for sodium, lead, or gas-cooled fast reactors, modular high temperature gas reactors, or high temperature gas reactors -- there is a set of sodium fast reactors that you see which was really based on the PRISM design. And there's a set of modular high temperature gas-cooled reactor design criteria which were based on a TRISO fueled, helium cooled, graphite moderated by a temperature gas reactor.

So the FMR design kind of fits between the SFR and mHTGR designs that were considered here. And as you'll see when I go through their design criteria, they mostly picked the advanced reactor design criteria, and then picked some mHTGR or SFR design criteria as the basis for their PDCs.

And I will highlight here that mixing and matching was explicitly considered in the development of this reg guide and endorsed by the reg guide. And there is an explicit quote that I'll read. "The applicants may use this reg guide to develop all or part of the PDC and are free to choose among the ARDC as an RDC, or mHTGR-DC, to develop each PDC after considering the underlying safety basis for the criterion and evaluating the rationale for the adaptation described in the reg guide."

And so that's something that was explicitly thought about as we were building this reg guide in the first place.

So General Atomics' approach to PDC development was generally to start with the advance reactor design criterion, consider the underlying safety basis. If the advance reactor design criterion wasn't fully applicable, they would then assess the more specific design criteria to see if they were acceptable to be adopted directly.

If they weren't directly applicable, they would find which one was most representative of the FMR design, and then they would adapt or refine whichever design criterion they went with as necessary. And we thought this was an acceptable approach.

And we were, while I did say on the last slide that, you know, the design kind of sits in a relatively comfortable space between the different sets of criteria that already listed, we did try to be mindful of whether additional criteria would be appropriate. And generally speaking, I think we were happy with where things were.

So now I'll spend a bunch more time on this slide talking about the effects of certain key design features on the principal design criteria. I didn't want to go through a detailed design overview, because John just did that for us.

So for the fuel they use, as John just mentioned, the uranium dioxide pellets and silicon carbide fuel pins, these are put into a triangular pitch and arranged into the hexagonal bundles that is typical of fast reactor design.

And this fuel design which, when you think about that coupled with the use of a leak-tight containment building, lends itself to the use of specified acceptable fuel design limits in the principal design criteria as currently exists for LWRs in general, Design Criteria 10.

Rather than the specified acceptable

system radionuclide release design limits, I think I got that right, that are, or SARDLs which I will say from now on, because it's much easier to say, that are in Reg Guide 1.232 for the modular high temperature gas reactor design criteria.

And we felt that that was appropriate for the fuel and core arrangement that they were considering rather than, you know, TRISO fuel, HTGR. And this affects the Criterion 10 and several other design criteria that reference back to Criterion 10.

The fast modular reactor is a fast reactor. So, as when you compared a thermal spectrum reactor, the core is more tightly coupled and more tightly coupled with the surrounding structure. So we wanted to make sure that the effects of structures on reactivity feedback would be considered. That's reflected in Design Criterion 11 and 12.

They use the helium coolant which affects quite a large number of principal design criteria. And consistent with the modular high temperature gas reactor design criteria, in the FMR design criteria they moved the emphasis kind of more from inventory control to ensuring that there's adequate residual heat removal.

There's no PDC 35 which is the ECCS design criterion. Also, throughout DC they've removed, uh, the term reactor coolant pressure boundary and replaced it with reactor helium pressure boundary. And some of their design criteria in this area related to the coolant adopt the modular high temperature gas reactor design criterion directly, while others adapt the advance reactor design criterion to fit the design.

The power conversion system is a direct Brayton cycle. They use the gas turbine that runs directly on the primary coolant so the turbine itself is therefore inside the reactor helium pressure boundary. And the overall power conversion system forms a portion of the reactor helium pressure boundary.

And that has to be considered in the environmental dynamic effects design basis. And General Atomics did that appropriately. They adopted a modular high temperature gas reactor, Design Criterion 4, which included those considerations in it.

The residual heat removal system, which was one of the last things John talked about, lends itself to using the MSGGR design criterion for passive residual heat removal, though I will note that there were a couple of tweaks there that I'll get to once I actually go through the criteria themselves.

And finally, the use of a leak-tight containment implies the use of all the regular containment principle design criteria. And for the containment design criterion itself, DC-16, the sodium fast reactor design criterion was used as the basis. And I'll talk about that in a little bit.

All right. So now I'm going to walk through the actual criteria themselves. So in this first set of requirements, the one that really stands out here is Criterion 4 which, as I just mentioned, means that the effects -- that they used the modular high temperature gas reactor design criteria which explicitly considers the effects of turbine missiles originating both inside and outside the reactor helium pressure boundary, which is very important considering the turbine is inside the reactor helium pressure boundary and wouldn't necessarily be considered otherwise.

In this block there's a bunch that I'm going to touch on. Criterion 10 is where they introduce the specified acceptable fuel design limits rather than SARDLs which, as I mentioned earlier, is appropriate for their fuel system in containment design.

It also replaces coolant in the design criterion with heat removal which is consistent with the modular high temperature gas reactor and other gas cooled reactors which argue that helium inventory control, specifically during a transient, isn't needed to meet the SAFDLs provided that heat removal is maintained.

Criterion 12 is based on the advanced reactor design criterion rather than the modular high temperature gas reactor design criterion because, as I mentioned before, structures have to be captured in the criterion due to the long mean free path of fast neutrons.

Also in adopting this criterion they've removed effects of coolant, because the coolant itself has a negligible impact on reactivity feedback. And that's consistent with the justification, and rationale, and the language in the modular high temperature gas reactor design criteria.

Criterion 16 is based on the sodium fast reactor Design Criteria 16 which includes the use of a low leakage pressure retaining containment concept. And I know I mentioned earlier that, in general, the reg guide endorses mixing and matching of design criteria. But this was explicitly discussed in the containment design basis criteria where it said, you know, we would expect that developers would use the criterion that best fits their containment design.

VICE CHAIR KIRCHNER: Reed, in this particular instance I would have thought they would have fallen back on the PDCs. This is a higher pressure containment.

And the sodium reactors are -- the containment there is to deal with leakage and fires, really, not pressure. Although you could have a pressure event if you mixed the water with the sodium because of a leak somewhere in the system and such. But I would have expected they, as I said, might have fallen back on the GDCs with regard to containment.

Also when you come to the later GDCs on containment testing for leak-tight integrity, Appendix J, etcetera.

MR. ANZALONE: Yeah. So thanks for bringing that up. I think there's a bunch of discussion in the reg guide about, you know, what is appropriate. And there were a bunch of Commission considerations, back starting in the '90s, about what would be appropriate for advanced reactor designs.

And I think the general consensus, if I remember correctly, was that the prescription of a, quote, unquote, "essentially leak-tight containment," which was what was in the GDC, was too stringent of a requirement for advanced reactor designs as opposed to the low leakage containment. And I thought that that was generically applicable.

And then the last one on this slide, I'm going to talk about is Criterion 17 which is the same as the modular high temperature gas reactor design criteria. And that's appropriate, but they've tweaked that to refer to SAFDLs instead of SARDLs.

MEMBER HALNON: Why would that be reversed from the previous. I thought the SARDLs was what they should be using.

MR. ANZALONE: No, they're using SAFDL.

MEMBER HALNON: They're using SAFDL?

MR. ANZALONE: Yeah. So they made it consistent throughout.

MEMBER HALNON: Okay. So SARDLs are more for the -- I think what you said there is the subcommittee was --

MR. ANZALONE: They're generally associated with TRISO fuel and --

(Simultaneous speaking.)

MR. ANZALONE: -- contained.

MEMBER HALNON: Okay. Thanks.

MR. ANZALONE: Any other questions? Sorry, I moved on.

Okay. Here I'm just going to, and I mentioned this during the subcommittee meeting, FMR-DC 26, we've had some challenges with PDC 26 with some applicants, but here General Atomics adopted the language in the AR-DC-26 as is, which we were pleased about, with one minor exception that they explicitly consider the effects of xenon as is originally included in the GDC.

And while we thought that xenon effects would likely be small for a fast reactor, during the subcommittee meeting General Atomics mentioned that they were indeed, they checked and found that they were small or negligible. We felt like it was conservative to include them. It doesn't hurt.

And then consistent with Reg Guide 1.232, the requirements of, or the design criteria in GDC 27 were incorporated into PDC 26. And that was done by General Atomics as well. So there is no Criterion 27.

And then for 28, even though the subject is reactivity limits, the most significant differences between the different sets of design criteria had to do with the coolant design. So General Atomics adopted the mHTGR-DC which fits the best for them there.

Fluid systems, and I kind of covered this in my overview, consistent with the modular high temperature gas reactor design criteria, there are no Criteria 33 or 35. Consistent with, you know, that removal of focus from inventory control to heat removal, or the residual heat removal criterion in 34, they adopted the modular high temperature gas reactor design criteria which, again, we thought was appropriate.

But they tweaked the title of it, and one or two of the words, to reflect that there are, rather than just a passive residual heat removal system, which was what was written into the Reg Guide 1.232 criterion, they wanted to encompass both the passive and active residual heat removal mechanisms that they had. And we felt like that was appropriate to include all of them within the scope. And that's reflected in 37 as well.

Next slide. These are all related to containment, and they just adopted the advanced reactor design criteria which we thought was appropriate.

More related to containment, so for 54, the sodium fast reactor design criterion was chosen, because it replaces the phrase, having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems in advanced reactor design Criterion 54 with -- 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.

So the intent of this change as described in the topical report is to accommodate designs that are capable of demonstrating that containment isolation valves aren't necessary for certain piping penetrations that don't have a credible release path to the atmosphere.

In the FMR design this includes the RVCS and intermediate power conversion system heat removal loops which could be designed to achieve the containment function without isolation valves. But if isolation valves, it turns out, are necessary, the design criterion still requires them to be included.

The other change here, which I have mentioned on the slide, is that FMR-DC 54 refers to containment rather than "reactor containment," quote, unquote, to reflect the presence of major SSCs containing radioactivity inside containment. So in this case, that's the power conversion system. And no comments on the ones related to fuel and reactivity control. They're all what you would expect to see.

So finally, the conclusions from our safety evaluation, we determined that General Atomics appropriately considered the reg guide and developed a sufficient set of principle design criteria that were appropriate for establishing requirements for the FMR design.

The PDCs themselves do meet that requirement of 10 CFR 50, Appendix A, to establish the necessary design fabrication, construction testing, and performance design criteria for safety-significant SSCs.

And as I mentioned, you know, there were a couple instances in which they expanded to ensure that all the safety significant, rather than just safety related, SSCs would be captured, and that the topical report could be used by future fast modular reactor applicants. But if the reactor design differs from that discussed in the topical report, use of the PDCs must be justified.

And that's just sort of part of the nature of how topical reports work in licensing space. You need to be able to say that the topical report is applicable. And that's the end of my presentation.

MEMBER BIER: Okay. Thank you. First of all, I appreciate that you kind of hit the high points and focused on the areas where there was something important or unique in the PDC and made it very efficient that way.

At this point, are there questions or comments for Reed?

MEMBER HALNON: Did you have any RAIs outside of the, you know, just minor questions?

MR. ANZALONE: So there were a couple of RAIs. I think most were related to, like, what I would consider to be errors in the PDCs, you know, grammatical errors --

MEMBER HALNON: Okay.

MR. ANZALONE: -- things like that. But there was at least one. So we asked about the inclusion of structures in PDC 12. So originally that wasn't in there. We felt like it was important to make sure that that was included so that the effect of structures on reactivity feedback would be considered. And they did duly add that back in.

MEMBER HALNON: Okay. Thanks.

MEMBER PETTI: So, Reed, how did you show completeness of the set, given this is, you know, a technology -- is between the couple technologies that have been looked at a lot. So that's always the question in the back of my mind, you know, is there something that's been missed?

MR. ANZALONE: Yeah, that's a good question. And it is, it's difficult to answer. Because I think it's something that we had in the back of our minds as trying to make sure that it was complete.

I think I will say that we did probably rely mostly on the fact that the criteria in the reg guide kind of encompassed, and I mentioned this at the subcommittee meeting, encompassed the scope of what we would expect to be considered for the design, you know.

So we're interpolating between designs effectively, right, between the sodium fast reactor and modular high temperature gas reactor designs. So I would say that's probably the true answer to the question. But there was some interrogation, you know, internally. Okay, have we captured everything? And I think that we did.

I will say if we are early enough in the design phase that if it turns out there is something that we somehow didn't include, it should come out later.

MEMBER PETTI: It's the same question I asked. So General Atomics, I mean, if you map it to the safety functions, and you think you've got all the safety functions which, I think, the LMP approach covers most of -- all the safety functions you would think of at a high level. It's sort of another way to check, but it's a cross check.

MR. ANZALONE: Yeah. Another project that I'm working on is using LMP, and it is very focused on identifying and then appropriately ensuring that you're considering those safety functions. So I agree that that would certainly capture anything that might have been missed.

Though, I mean, really the safety functions for this are not too different from what you would expect to see, again, from either the modular high temperature gas reactor or the sodium cooled fast reactor. So I don't see -- and I think, again, this was something that I said at the subcommittee meeting, you know, it's not really exotic. So I don't see that there's some area, obviously at least, that we would have missed.

MEMBER BIER: Yeah.

MR. ANZALONE: Of course, if it was obvious, then we would have considered it.

(Laughter.)

MEMBER BIER: Just following up on Dave's point, I don't think, you know, I'm not aware of a specific concern here. But in principle with this kind of mix and match thing, you can have interaction effects where a feature from one design doesn't marry happily with some --

MR. ANZALONE: Yeah. And certainly that's something we were very focused on during the review, is that if they were mixing and matching, was it appropriate to do so, and did they make sure with their mixing and matching that they were consistent. And that I can say very confidently, yes, we did make sure that that wasn't an issue.

CHAIR REMPE: Oh, I'm kind of going from Part 53 discussions, and it's not focused on this application. But I thought with Framework B there was discussion that well, yeah, it might be easier if one used the LMP approach. But we're going -- if people want to do a bottoms up approach, we'll let them, but we'll group things.

Now, sounds like General Atomics sort of did start with the critical safety functions, as we asked them earlier today. But are you guys grouping them like we've heard that might occur? I mean, this was done -- we don't have Part 53 in place yet, but is that kind of what staff's going to be doing as the try and deal with a lot of these more different designs that may come in?

MR. ANZALONE: I don't know that I have a great answer to that question, because I haven't started thinking about it in terms of people who aren't pursuing an LMP type framework. Because General Atomics is planning to use LMP. The other reactor designs that I'm working with are using LMP. So I don't know that I personally have an answer --

CHAIR REMPE: I just was --

MR. ANZALONE: -- to that question.

CHAIR REMPE: Yeah.

VICE CHAIR KIRCHNER: My fall back answer on your behalf would be that the GDCs are organized according to safety functions, right?

MR. ANZALONE: Yeah. Yeah, they are.

VICE CHAIR KIRCHNER: It's not explicit as LMP approach, but the categories that they're binned in makes sense from a safety function standpoint.

MR. ANZALONE: Right. And when John was asked that question, what came in my head was, well, they kind of tell you what the safety function they're related to is in the title or at least the heading of the sections.

So that's part of the answer. But don't feel like that's the whole answer to your question.

CHAIR REMPE: Too early to really have an answer. I'm just --

MR. ANZALONE: Yeah.

CHAIR REMPE: -- thinking ahead on how we're going to deal with it.

MR. ANZALONE: It's a good question though for framework lead.

CHAIR REMPE: And we had time in the schedule to explore it, so thank you.

MEMBER PETTI: I think though it is worth a letter at least touching on this, that the staff was very focused on the mix and match and making sure that something wasn't missed. Because we're going to see more mix and match, I'm sure.

MEMBER BIER: I noticed you mentioned no individual feature was that exotic. It's more the mix and match that could create issues.

Additional questions or comments for Reed?

MEMBER DIMITRIJEVIC: I just want to point out that in the reg guide actually the both sodium and, you know, module that have extended GDCs, you know, to additional ones in the '70s. Here, I mean, it seems like they're just, you know, keeping it simple to the basics.

MR. ANZALONE: Yeah. And I can comment on that a little bit. So the extended ones for the modular high temperature gas reactor relate to sort of the characteristics of the reactor building that need to be there when you use a functional containment. And the ones for the sodium fast reactor relate specifically to sodium and what nasty things that can do when it interacts with the atmosphere.

So neither of those are really considerations here since they're not using a functional containment approach, and they don't have sodium coolant.

MEMBER DIMITRIJEVIC: Well, that's very true. I just want to say maybe, you know, when they really go to the detailed design they will find some features not covered. I mean, we cannot really guarantee completeness about they have, you know, they have not made the radical changes and they have not extended. It's sort of basic application.

MEMBER BIER: Are there questions or comments?

Yes? Yes?

So if there are no questions in the room or from the members online, then I think it's time for any public comments on the GA principal design.

And I guess it's Star 6 to unmute yourself if you're on the phone.

And hearing none, I think we can close public comments.

CHAIR REMPE: Great. So at this time we're going to go off the record. And this is it for the court reporter for this meeting, okay.

(Whereupon, the above-entitled matter went off the record at 2:07 p.m.)

General Atomics – Electromagnetic Systems Fast Modular Reactor Principal Design Criteria

Reed Anzalone, NRR/DANU

Samuel Cuadrado de Jesus, NRR/DANU

Sheila Ray, NRR/DEX

Steve Jones, NRR/DANU

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Agenda

- 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

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**."

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	Consider in environmental and dynamic effects design basis
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.

GA-EMS FMR Design Features

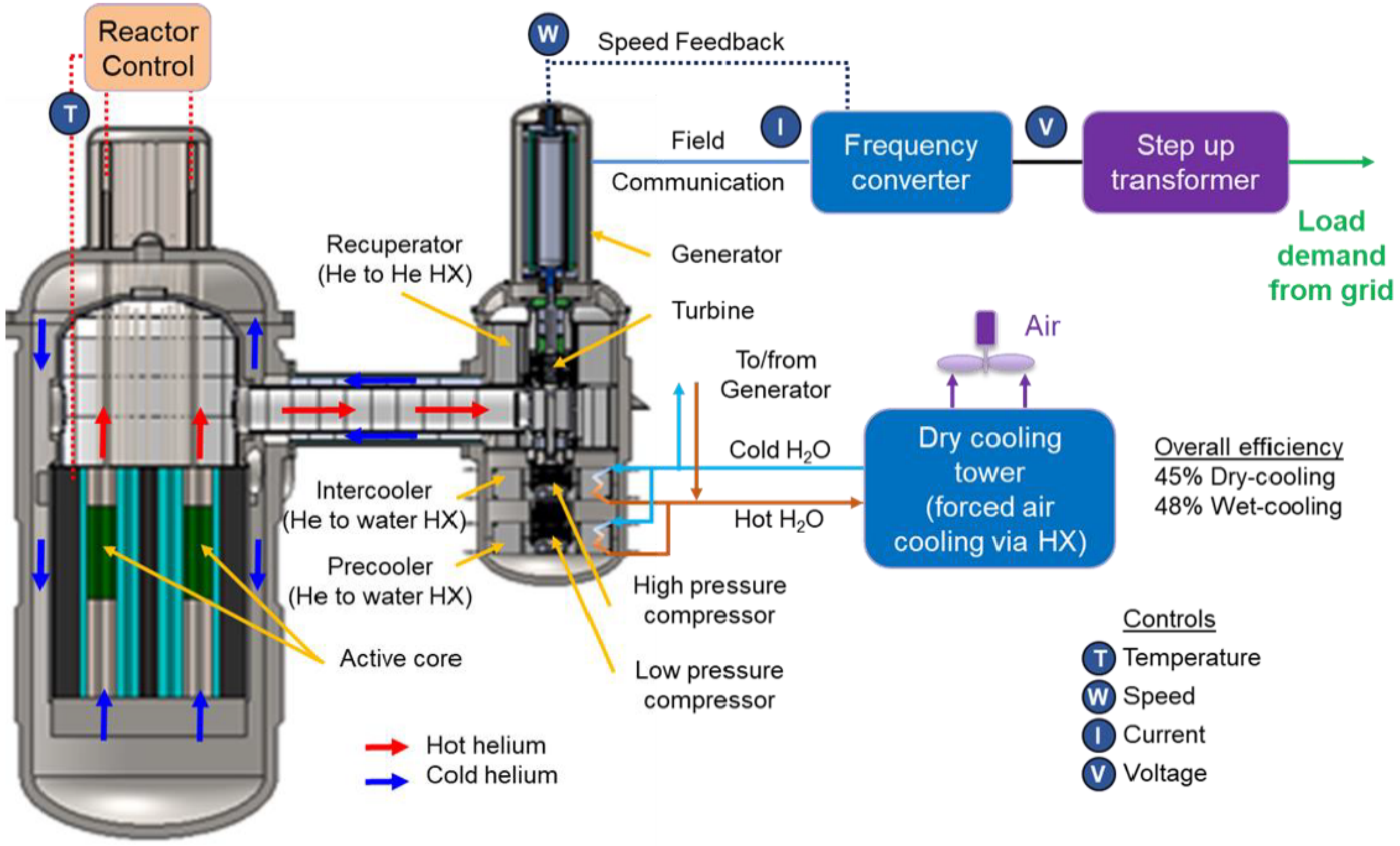

Figure 2. FMR Primary Coolant Flow and Power Generation

Source: REP, ML22087A510

GA-EMS FMR Design Features

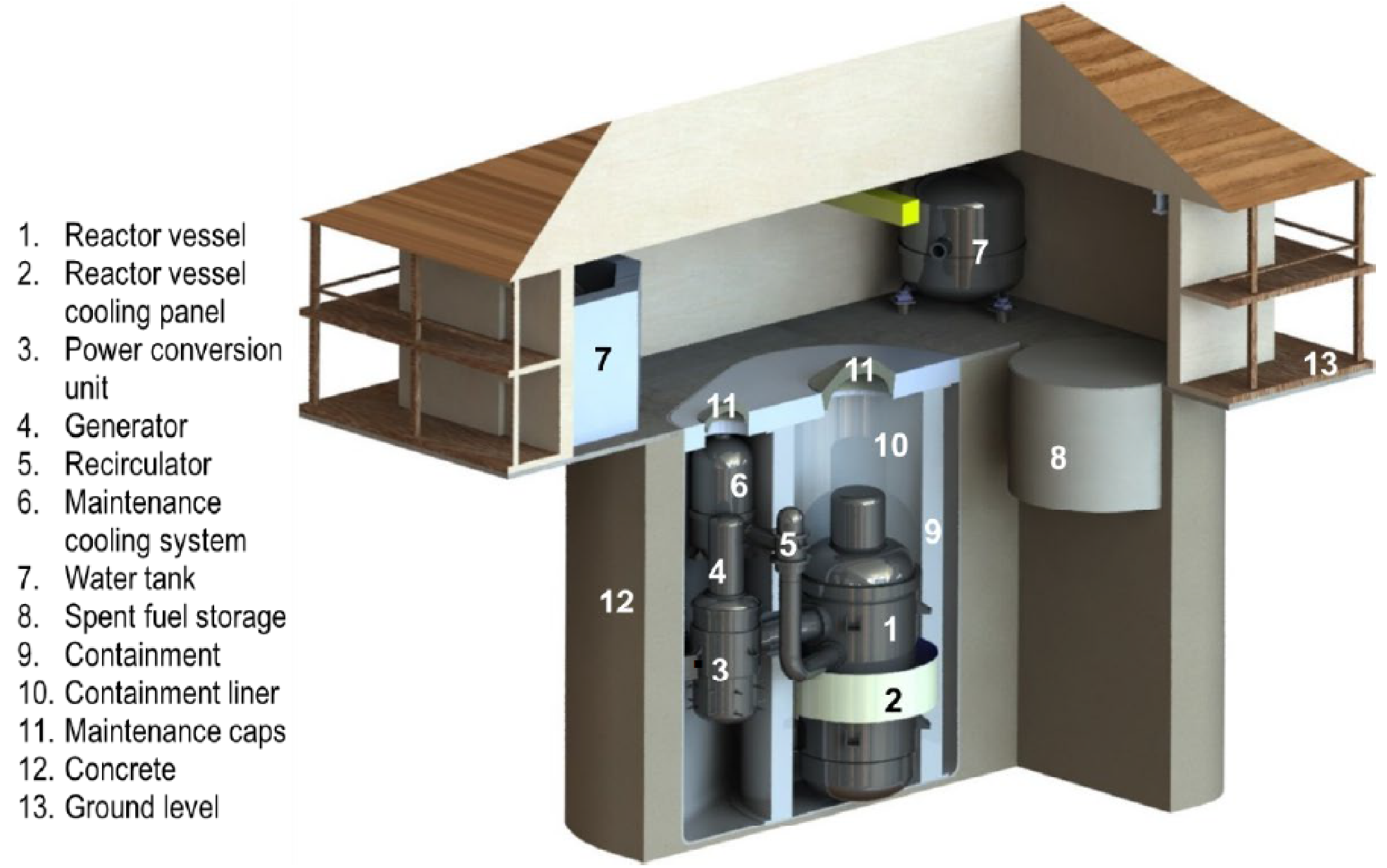

Figure 1. FMR Nuclear Island Components

Source: TR, ML22154A556

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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

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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

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FMR-DC Modified from RG 1.232

MHTGR-DC 34	FMR-DC 34
Passive 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. During postulated accidents, the system safety function shall provide effective 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.	~~Passive r~~Residual heat removal. ~~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 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.

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).

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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

General Atomics Electromagnetic Systems

Fast Modular Reactor Conceptual Design

June 7, 2023

Presented To: Advisory Committee on Reactor Safeguards

Prepared By: John Bolin (GA-EMS)

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

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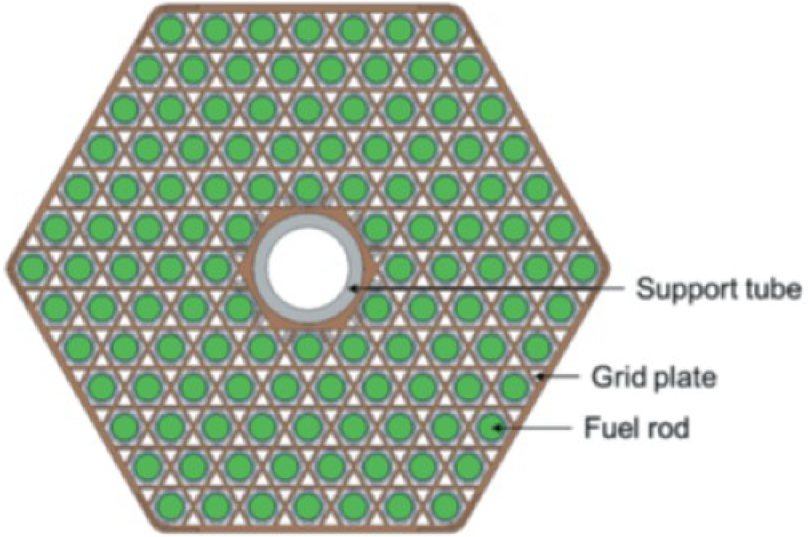
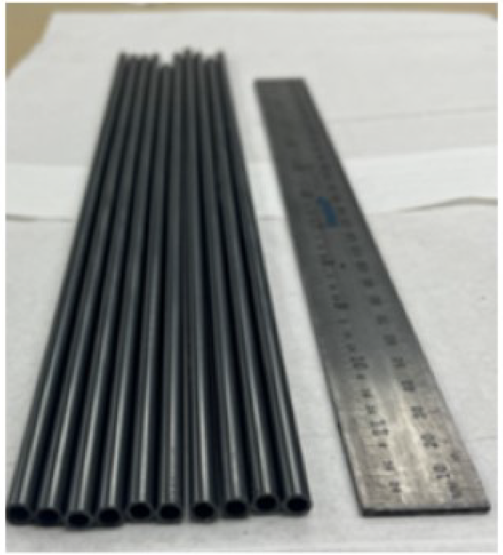
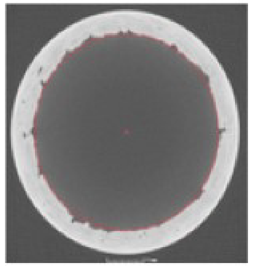
Fuel Leverages UO_2 Legacy and SiGA™ Cladding Development

- 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

Scaling analysis

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

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

Standard rodlet

Compact rodlet

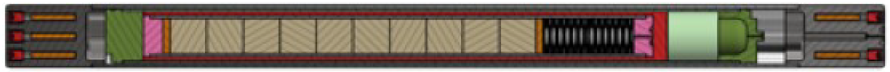
Documents

INL
- Capsule design and analysis
- Assembling

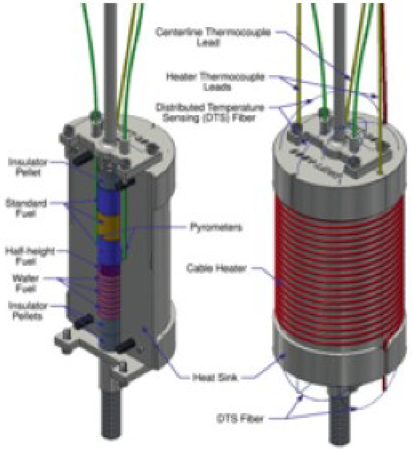
ATR irradiation

TREAT tests

PIE
- Fission gas release
- Deformation
- Fracture



ATR Irradiation Capsule



TREAT Transient Capsule

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FMR Test Rodlets Fabricated Using ATF Established Procedures

Fuel Rod Fabrication Process

Uranium Dissolution Precipitation

ADU

Mixing

Granulation

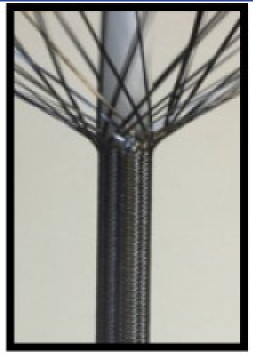
Pelletizing

Sintering

Grinding

QC

Fiber Braiding



Composite Fabrication

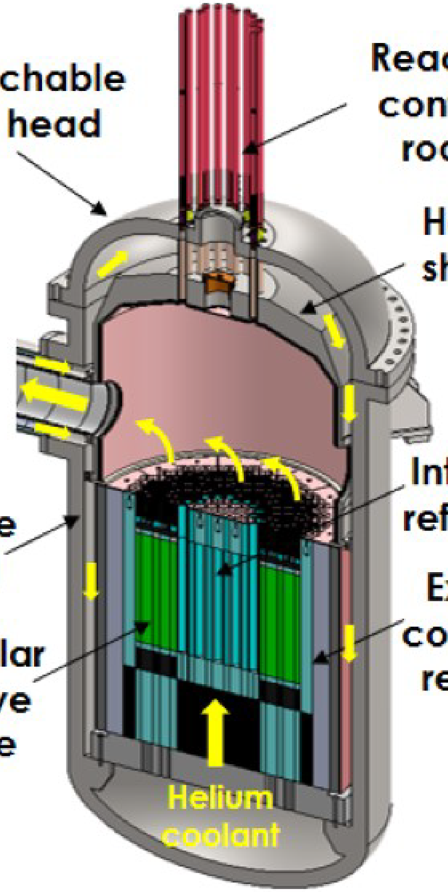
Dimensional Control

Pellet Loading

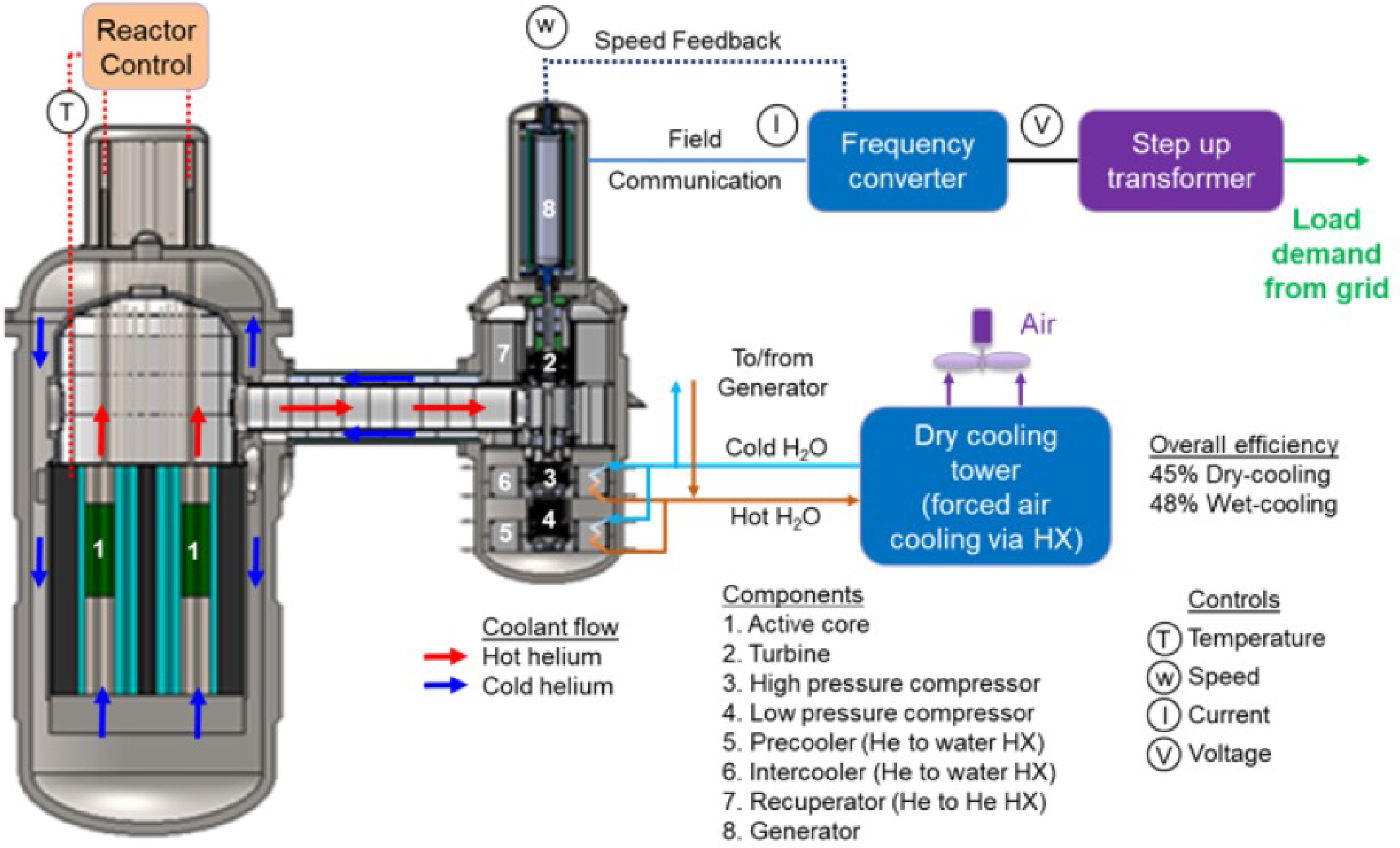
Joining/ Sealing

Vessel System Designed to Minimize Helium Leakage

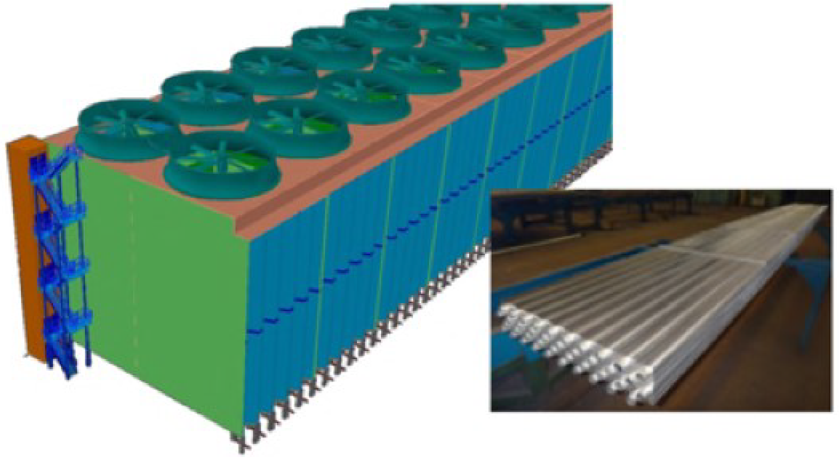
- **Conceptual sizing calculation for normal and AOO conditions**
  - Design code is Section III, Division 5, 2021 Ed.
  - Thickness is adequate for operation up to 300,000 hours (~34 EFPY), will be extended to 540,000 hours (~60+ EFPY) (code revision)
  - Proven use of seal welds at joints to minimize 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

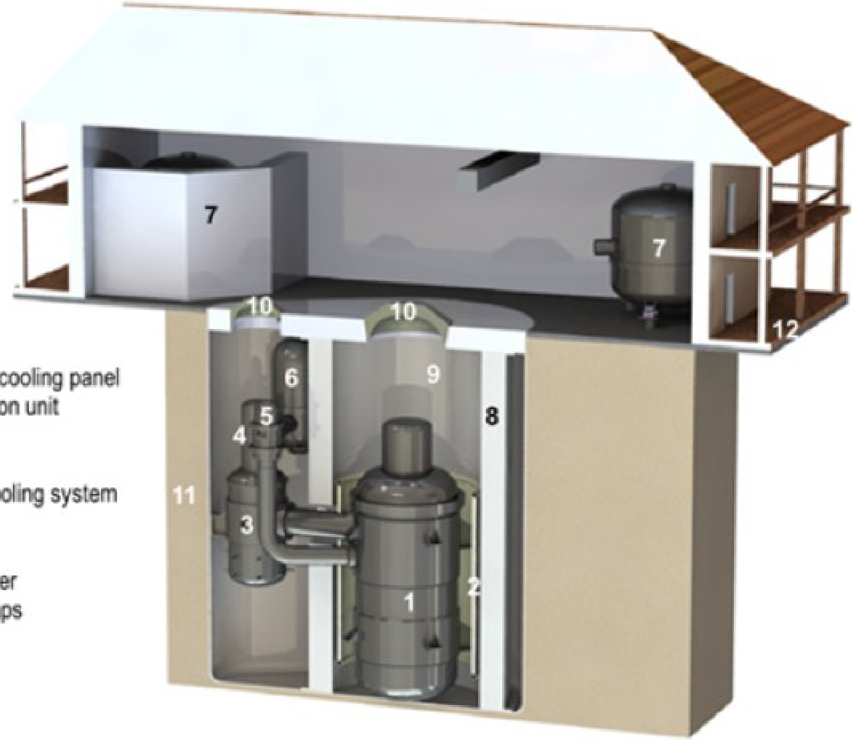


- **Dry Cooling Tower**
  - Reduces impact on water resources and expands siting options



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 is less vulnerable to airplane crashes

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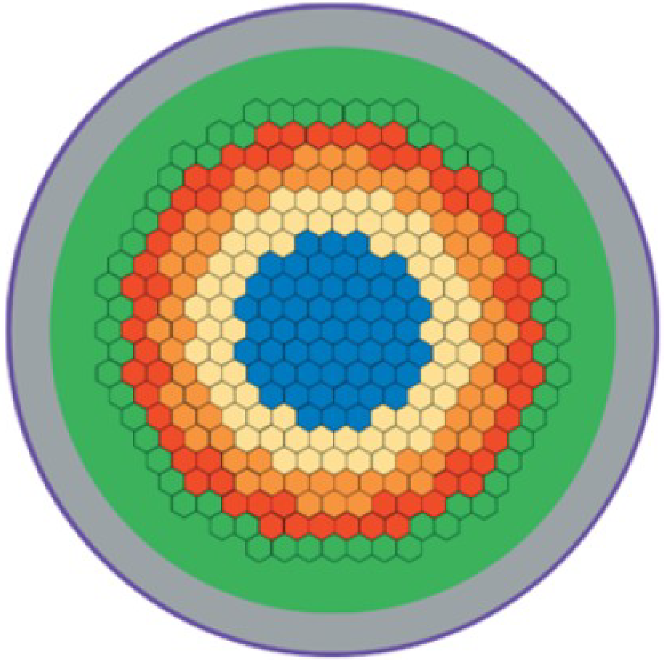
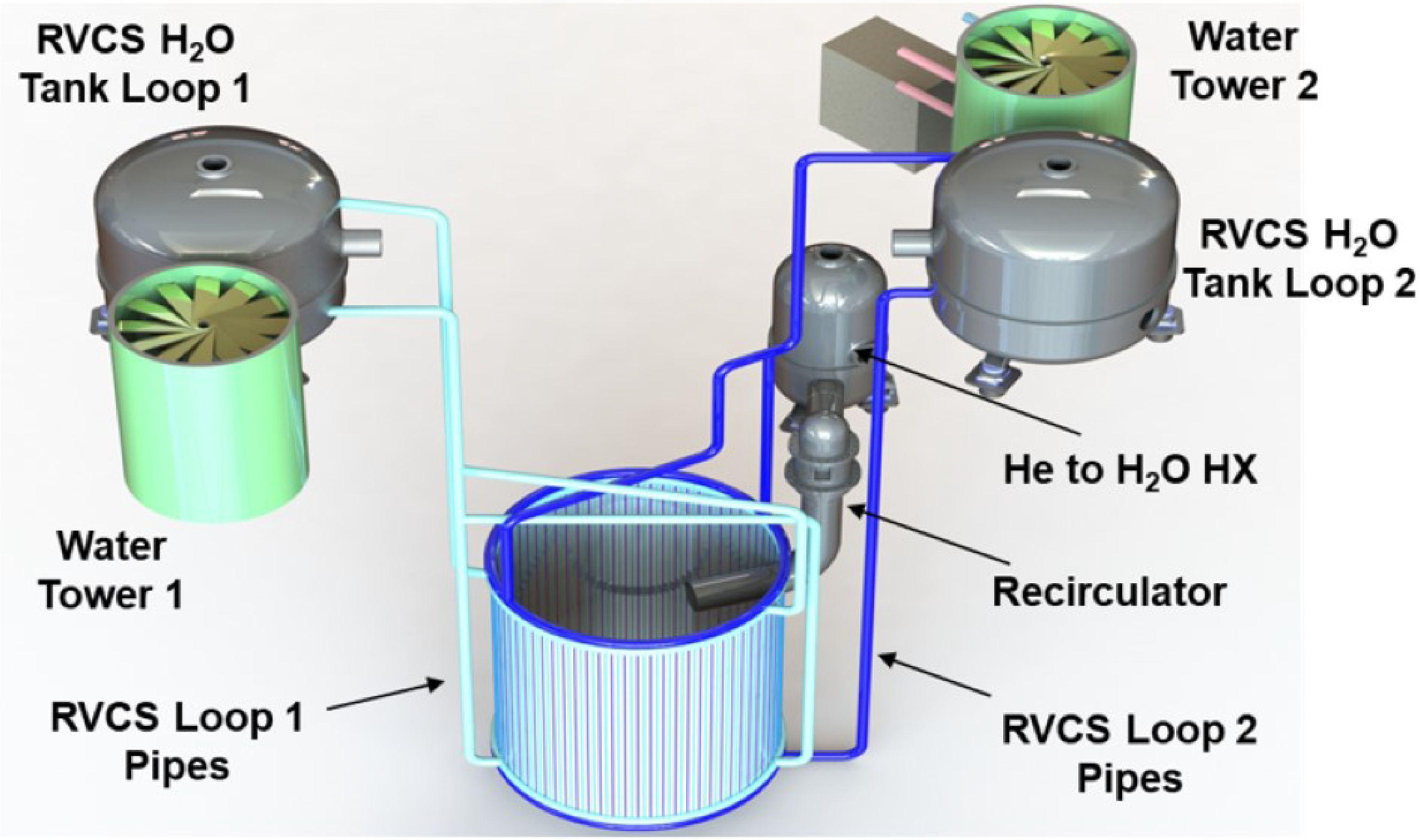
Residual Heat Removed By Active and Passive Systems

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





FMR core and RVCS design guarantees long-term heat removal during severe accidents

Acknowledgements

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