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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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KAIROS POWER SUBCOMMITTEE

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THURSDAY

APRIL 21, 2022

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The Subcommittee met via Video-Teleconference, at 1:00 p.m. EDT, David A. Petti, Chairman, presiding.

COMMITTEE MEMBERS:

DAVID A. PETTI, Chairman

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

CHARLES H. BROWN, JR. Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

JOSE MARCH-LEUBA, Member

WALTER L. KIRCHNER, Member

JOY L. REMPE, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANTS:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

A-G-E-N-D-A

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

1:00 p.m.

CHAIRMAN PETTI: The meeting will now come to order. This is a meeting of the Kairos Power Licensing Subcommittee of the Advisory Committee on Reactor Safeguards. I'm David Petti, Chairman of today's Subcommittee meeting.

ACRS members in attendance are Jose March-Leuba, Joy Rempe, Matt Sunseri, Walt Kirchner, Vesna Dimitrijevic, Vicki Bier, and Greg Halnon.

Ron, are you on? Yes, I see Ron Ballinger. And I haven't seen Walt Kirchner yet.

MEMBER BROWN: I'm here, Dave.

CHAIRMAN PETTI: Oh, and Charlie, you are there. Okay, good.

MEMBER BROWN: Yes.

CHAIRMAN PETTI: Thank you. Steve Schultz and Dennis Bley, our consultants, are also present. Weidong Wang of the ACRS staff is the Designated Federal Official for the meeting.

During today's meeting the Subcommittee will get an overview of the Kairos Hermes Testing Facility construction permit application and the NRC staff approach to the safety review. The Subcommittee will hear presentations by and hold discussions with

NRC staff, Kairos Power representatives, and other interested persons regarding this matter.

The part of the presentations by the Applicant and the NRC staff may be closed in order to discuss information that is proprietary to the Licensee and its contractors pursuant to 5 U.S.C. 552(b)(c)(4).

Attendance at the meeting that deals with such information will be limited to the NRC staff and its consultants, Kairos Power, 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 the closed part of the meeting.

The rules for participation in all ACRS meetings, including today's, were announced in the Federal Register on June 13th, 2019. The ACRS section of the U.S. NRC public website provides our charter, bylaws, agendas, letter reports, and full transcripts of all full and subcommittee meetings, including slides presented there.

The meeting notice and agenda for the meeting were posted there. We have received no witness statements or requests to make an oral

statement from 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 in the future.

A phone bridge line has been opened to allow members of the public to listen in on the presentations and the Committee discussion. Additionally, we've made an MS Teams link available, and there will be an opportunity for comment at the conclusion of the prepared presentations for the public that is interested in making such a comment.

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

Additionally, participants should mute themselves when not speaking. To mute or unmute on a phone, push Star 6. If you're on Teams and want to make a public comment, you can just raise your hand when we ask for public comments.

We'll now proceed with the meeting. And I'd like to start by calling up NRR management, Duke Kennedy. I understand you're going to say a few words.

MR. KENNEDY: Good afternoon, everybody, members of the ACRS, members of the public in attendance. And I'd just like to give a few opening remarks. So my name is Duke Kennedy, and I am the acting chief of the Advanced Reactor Licensing Branch. And it's my pleasure to be here today to provide introductory remarks on behalf of the Division of Advanced Reactors and Non-power Production and Utilization Facilities in the Office of Nuclear Reactor Regulation.

With me today is Mr. Ed Helvenston of the Non-Power Production and Utilization Facility Licensing Branch who is one of the project managers for the Hermes review. And he'll provide the staff presentation.

Also here is Jeff Schmidt of the Advanced Reactor Technical Branch, who's the lead technical reviewer, as well as Ms. Michelle Hart, another technical reviewer, and other NRC staff who are involved in the Hermes review.

So I'd like to thank the ACRS Subcommittee for convening this meeting today to provide the staff an opportunity to introduce the ACRS to the staff's approach to the Hermes review.

So we recognize that Hermes represents the

confluence of two uncommon attributes and that it's a non-light-water reactor design, and it's proposed to be licensed and operated as a testing facility under Section 104(c) of the Atomic Energy Act of 1954.

So recognizing this unique, novel situation, the staff is pursuing a deliberate risk-informed approach to this review with a focus on safety and reasonable assurance of protection of public health and safety.

So the staff and Kairos Power have had the opportunity to brief the ACRS on Kairos Power topical reports. Some of these are applicable to both the power reactor design and the non-power Hermes design. So the staff has appreciated helpful comments from the ACRS on topical reports covering different areas such as reactor coolant, scaling methodology, the licensing modernization project, and more recently fuel qualification and mechanistic source term. So many of these topical reports are or will be referenced in the Hermes application.

So the staff looks forward to continued interactions with the ACRS Subcommittee as this review proceeds. Of course, we're at the initial meeting here to kick things off and look forward to hearing presentations from Kairos and questions and comments

from the ACRS members as well. So thank you very much. I'm looking forward to an informative meeting.

CHAIRMAN PETTI: Thank you, Duke. Before we turn it over to Kairos, just to reiterate for members, we do have a closed session if we want to get into more technical details, so just note that.

Kairos, the ball's in your court.

MEMBER REMPE: Dave, this is Joy.

CHAIRMAN PETTI: Yes?

MEMBER REMPE: Could I ask a question of Duke before we go to Kairos?

CHAIRMAN PETTI: Okay.

MEMBER REMPE: Are you still there? Oh, okay. So I've been pondering the last few weeks here about what the NRC does to decide whether a facility is a test reactor or a demonstration reactor. Because we hear about, well, it's a lighter footprint with a test reactor with respect to licensee.

And if I actually read, like, in Section 10 of the DCA, it says that the test reactor is being constructed to demonstrate this new technology and there aren't any special test facilities. You'll do startup testing, but we do startup testing for commercial reactors in some respects.

How does the NRC decide? And then what's

the cutoff with respect to power levels where you suddenly say, no, come on guys, this is not going to be a test reactor? It's a demonstration facility, and you have a power level that shouldn't go through a certain type of process. Are there some hard and fast rules?

MR. KENNEDY: Well, Ed Helvenston will touch on some of that in his presentation. But in response to your questions, first, the term demonstration reactor actually has a fairly particular meaning when it comes to the Atomic Energy Act. And it typically includes connection to an electrical grid and demonstrating that the reactor technology can be commercialized.

So Kairos Hermes will not be connected to an electrical grid. They will not produce electricity, and therefore wouldn't fall under that definition of -- it's not a definition, but the classification as a demonstration reactor per the Atomic Energy Act. So it is a research and development facility.

It is not a demonstration reactor, although it may be used to demonstrate some of the technologies or safety features. It doesn't fit under those clauses in Section 202 of the Energy

Reorganization Act that talk about demonstration reactors. So I think that's the answer to your first question.

To the second question, there is a few criteria that determine whether a Class 104(c) research and development facility is a research reactor or a testing facility. The easiest one is the ten-megawatt thermal power cutoff limit. Above ten megawatts, a facility is a testing facility. Below ten megawatts, it would be research reactor.

So at or below ten megawatts it would be research reactor unless it meets certain conditions that are laid out in 10CFR Part 50. And those relate to other features such as a liquid fuel, Kairos does not have a liquid fuel, or a large cross sectional area in the core that could be used for experiments.

And so these features would be restricted to reactors with a power level of one megawatt. So if these features, none of these features existed, and the reactor power was greater than one megawatt, it would also be classified as a testing facility.

So Kairos, being greater than ten megawatts thermal, the Hermes reactor, it is clearly a testing facility, or that's where it would fall under section 104(c) of the Atomic Energy Act.

There is no actual upper bound to power level in the Act or the regulations. The designation of testing facility is based on the facility being useful for research and development. So if it were 100 megawatts or 200 megawatts, it would still be eligible be classified as a testing facility under the Act and the regulations.

The next categorization would be a commercial facility, and that would be dependent upon the types of activities that it's carrying out. And Mr. Helvenston will explain this more in his part of the presentation.

So the answer is there is not an upper bound, and the staff recognizes this, and the guidance recognizes this. And so we are prepared to be able to apply our review at the right level considering the risks of the facility which can increase as power level increases. And so we have the flexibility to treat this case with the due diligence needed respecting the potential risks.

MEMBER REMPE: Thank you. That helps a lot. Have you had any experience in the past of ever applying the regulation to a testing facility of this magnitude? I know NRC was involved in the FFTF approach, but have you ever had the responsibility to

license a testing facility of this power level?

MR. KENNEDY: Not that I'm aware of. As far as I know, the National Institute of Standards and Technology reactor at 20 megawatts is the closest that NRC has licensed. If I'm incorrect about that, I will provide that information to the Subcommittee. But to the extent of my knowledge, we have not.

MEMBER REMPE: This helps. Thank you very much.

MR. KENNEDY: Thank you.

CHAIRMAN PETTI: Okay, Kairos.

MR. PEBBLES: All right, thank you, Mr. Chairman and members of the ACRS. My name is Drew Pebbles, and I'm a licensing manager here at Kairos Power.

As Duke mentioned, we have the opportunity to engage the Subcommittee on several of our topical reports that we submitted in pre-submittal phase. And we look forward to engaging as you begin your review of the Hermes PSAR.

Before we get started on the presentation, I did want to provide some context for the level of detail that you can expect in this presentation as well as the level of detail that you can expect in the PSAR that you'll begin to review.

First it's worth noting that we are following the Part 50 process which is a two-step process. And the construction permit application is based on a preliminary design and a preliminary safety analysis report.

Because of that, you won't see the same fidelity in design or the safety case that you could expect to see with the operating license application. And today's overview reflects some of that level of detail.

Second, it's worth noting that Hermes is a non-power reactor. So the requirements in Part 50 are slightly different for non-power reactors than they are for power reactors. We recently got approval of our topical report with KP-TR-004, which is a regulatory analysis topical in which we broke down all of the requirements in Part 50 and which ones apply and do not apply to the Hermes reactor.

And then finally, just due to the past constraints of today's meeting, we won't be able to go into detail in every system that you would expect to see in the PSAR. But we tried to pick the major and most important structure systems and components that if you do have any specific questions on supporting systems that you don't see in the presentation feel

free to ask.

Michael, are you sharing your screen?

MEMBER REMPE: Dave, this is Joy. I have another question. I can't raise my hand very easily. It was okay to ask it?

CHAIRMAN PETTI: Sure.

MEMBER REMPE: While you're getting the slides up, I became aware of last week that you guys had submitted some updates to your construction permit where, you know, there were some substantial changes, like you eliminated an intermediate cooling loop. And you changed the operating number of years from ten to four years.

Could you talk a little bit about what made those substantial changes, you know, what motivated you to make such changes? And should we expect similar changes coming down the pike here? Because, you know, we have limited time to do this review.

And I know Applicants often complain about how much it costs to go through an NRC review. And when you're making that kind of change, that increases costs. And so it'd be good to have some confidence that we are expending our review time at the right time.

MR. PEBBLES: Sure. Thank you for that question. As far as the getting rid of the intermediate loop, we don't think it had a materially large impact on the application. We were in very close contact with the NRC reviewers as we made the change.

It turns out that there is very little safety significance to that part of the plant. So where it showed up in the application was actually relatively minor compared to some of the other systems that play a more important role in the safety case for Hermes.

As far as the operating life, that came about from the NRC review of some of the associated topical reports that are currently under review with the NRC. And it turned out, in the case of operating life, that it could have slowed down our development path which I'll talk about a little bit in the introduction slide. But again, this wasn't a large material change to the application.

CHAIRMAN PETTI: Just a follow on question on the lifetime, you know, pebble beds can take a fair amount of time to get to equilibrium. Is it four full-power years, or four calendar years? And how long, you know, relative to when you're going to get

to equilibrium, how much are you projecting in terms of operating beyond equilibrium?

MEMBER MARCH-LEUBA: Somebody answering?

MR. PEBBLES: So it is calendar years. I'm going to ask one of my subject matter experts for the timing to equilibrium.

MR. SATVAT: It depends, it's close to a year.

CHAIRMAN PETTI: So introduce yourself.

MR. SATVAT: This is Nader Satvat, manager of Core Design. The residence time of Hermes reactor is about close to 200 days. So if the reactor operates steadily, it will get to equilibrium in about a year.

CHAIRMAN PETTI: Yeah, but when you start up, are you starting up like a traditional pebble-bed where you start out with graphite pebbles and slowly add lower image pebbles and then, you know, as those burn, add higher image pebbles? So there's this period where there's a lot of stuff going on with, let's say, but not the steady state fuel element, if you will.

MR. SATVAT: It's an area that we are studying. But effectively, your assessment is correct. We will adjust the effective abridgement of

the core to stay within a reasonably small excess reactivity, whether using solar enriched fuel or additional natural uranium to the core at the startup, to bring the effective enrichment down.

CHAIRMAN PETTI: Okay. It's just, you know, you'd like to get a -- I'm assuming you guys want a fair number of pebbles to get to full burn-up in the four years. That would seem to be a very good goal.

MEMBER REMPE: So I heard answers to my first two comments about the changes and that you viewed them to not be significant. I didn't hear about are we going to see some additional changes in the construction permit, or you think it's fairly stable here?

MR. PEEBLES: We think it's stable. There may be minor changes that result from the discussions with the review staff as we get through the current audits that are open and any requests for additional information that could come from the staff. But we are not planning any major changes to the construction permit application. Yes, sorry.

CHAIRMAN PETTI: Okay, keep going. If you hear silence march forward.

(Laughter.)

MR. PEEBLES: Okay, next slide please. All right, Kairos is a very mission-driven company, so we like to start every presentation by reiterating our mission which is to enable the world's transition to clean energy, with the ultimate goal of dramatically improving people's quality of life while protecting the environment.

So, in order to achieve this mission, we have to prioritize our efforts to focus on our clean energy technology, specifically the KP-FHR, and make sure that it is affordable and safe.

Next slide, please. So a quick look at the agenda. I'll give a brief introduction to Kairos and where the Hermes reactor fits in our development path.

And then I'll turn it over to the technical team to discuss the fuel and core design, the reactor vessel and internals, the heat transport systems, including the normal primary heat transport system and the safety-related secure heat removal system, as well as the pebble handling and storage system.

Then we'll talk about some of the safety-related structures like the reactor building, the I&C and electrical, and then we'll follow-up with an

overview of the safety case.

Next slide. Oh, delay, okay.

So a little bit about Kairos Power. Like I said on the mission statement, we're singularly focused on commercializing our clean energy technology which is the fluoride, salt-cooled, high-temperature reactor, or FHR. We were founded back in 2016, and we're at a current staffing level of about 269.

That number is probably already out of date, because we're growing every day. And it's also worth noting that 90 percent of that staff is engineering-focused which just underscores how committed we are to achieving our mission.

We're privately funded, and our schedule is driven by a goal to commercially demonstrate by the 2030s. That target date is based on when a large capacity of natural gas is expected to retire. So our cost targets are also in line with those natural gas plants.

In order to meet those aggressive costs and schedule goals, we've adopted a rapid iteration approach to developing our technology. We use rapid iteration throughout our development process, but on this slide we depicted several of the major hardware milestones that will occur from these iterations.

So if we start over on the left part of the slide, you see the engineering test unit demonstration experiment which is a non-nuclear water-based system that's up and running here in our facility in Alameda.

Next is the engineering test unit which is a non-nuclear Flibe-based system. It's a scaled down version of our commercial reactor. And the scale is actually very close to the Hermes reactor. It's in the final stages of being completed and should be operational within the next couple of months. And that is located at our facility in Albuquerque.

We will be able to incorporate a lot of that learning into the next iteration, which is our first nuclear demonstration, which is the Hermes reactor that I'll talk a little bit about on the next slide.

And then following the Hermes reactor, we have a full scale version of the commercial plant that's non-nuclear that will be used for user training and other purposes, that's our U-facility, and then finally, the first commercial plant.

Next slide.

MEMBER REMPE: This is Joy. I had a question or a comment on the past slide. When you go

from the 35 megawatts thermal up to the 100 and, well, when you go up from the Hermes reactor to the 140 megawatt electric plant, how do you know that the Hermes is going to be of sufficient scale that you'll have confidence in the commercial plant?

I mean, we've got a long history in the US, as well as Germany when they went from AVR to THTR. We went from Peach Bottom to the Fort St. Vrain reactor. And scale up led to problems that the larger plants weren't commercially viable. What gives you confidence you've captured enough of the salient features in the Hermes that your scale-up's going to work?

MR. HAUGH: Thanks, Joy, this is Brandon Haugh, Director of Modeling and Simulation. Classically when you look at the LWR fleet especially, I think you picked some that were, you know, gas reactor types that have their own challenges.

You know, this type of scale above 10-X is very common. They went from very small, to medium, to large. And large is very large. We're not making those kind of leaps. So we figured this 10-X step is very reasonable compared to previous technologies.

And also the safety features, and behaviors, and systems are pretty much identical at

the Hermes reactor which has some small changes in scale. Particularly around safety systems and safety features, they're nearly identical. So that's the reason we think that step is not too big, and it doesn't present any undue risk.

MEMBER REMPE: You're telling me you think that the molten salt reactor is more similar to a light-water reactor than a non-LWR type of scale-up, huh?

MR. HAUGH: No, I'm not saying that, I'm just saying there is precedents. And in the confidence in our technology, we believe that's a reasonable step.

MEMBER REMPE: And that confidence comes from the molten salt reactor experiment at Oak Ridge or --

MR. HAUGH: It comes from a combination of all the technology development activities we're doing, and the safety case we're presenting, along with our whole reactor program.

MEMBER REMPE: Thank you.

MR. HAUGH: Thanks.

CHAIRMAN PETTI: Just a quick clarification that the U-facility will be the same power as the commercial?

MR. HAUGH: The U-facility is an electrically heated facility to demonstrate the full scale primary system and to help with training on operators and maintenance. So the electrical power level is not determined yet, because it won't be there. It's not to produce power, and it's --

CHAIRMAN PETTI: Right. But thermally, in terms of heat fluxes, you're going to try to match, you know, those sorts of things?

MR. HAUGH: We haven't determined that yet. Because it's not necessarily a facility that tests in terms of safety and things. It's more to demonstrate the physical capability to manufacture it and also to train people to work on the full scale equipment.

CHAIRMAN PETTI: Okay. So it could be a step between the 35 megawatt Hermes and the commercial?

MR. HAUGH: Yes. I would very much expect that the electrical load we put in to heat the U-facility is much smaller than a commercial and nuclear heat load we would have in the KPX reactor.

CHAIRMAN PETTI: Okay. Thanks.

MEMBER REMPE: And the --

CHAIRMAN PETTI: That was Brandon Haugh,

the director of Modeling and Simulation.

MEMBER REMPE: And, Brandon, I guess I just have one final comment, I think, about the steam generators at San Onofre when you talk about the confidence in scale-up with the light-water reactor industry.

MR. HAUGH: Well, there's a whole other set of reports on what went wrong there. And that has nothing to do --- well, it had something to do with the scale-up, but a lot of other things, I don't think, are comparable.

This is Brandon Haugh again, I used to work there, so --

MEMBER REMPE: I know.

MR. PEEBLES: All right, so on this slide, just a little more about Hermes. The figure on the right gives you an idea of scale between both the non-nuclear ETU, and the nuclear Hermes, and the non-nuclear U-facility and KPX.

So what are we trying to demonstrate with this reactor? First and foremost, cost, establishing a competitive cost through our iterative learning cycle, which is part of a deliberate and incremental risk reduction of the design and testing iteration loops.

We're also flexing the supply chain which also has an effect on cost, but making sure that we're advancing the supply chains for specialized KP-FHR components and materials. The licensing approach, although the non-power reactor licensing approach will be slightly different, licensing certain safety concepts with Hermes will help inform the licensing process for the KPX reactor.

And then finally, operations, providing a complete demonstration of nuclear functions, including reactor physics, fuel, structural materials, irradiation, an radiological controls.

MEMBER REMPE: This is Joy ---

CHAIRMAN PETTI: So just a ---

MEMBER REMPE: Oh, go ahead Dave.

CHAIRMAN PETTI: Just, I would imagine, you don't actually say it on the slide, but I would imagine that any sort of specifications and procedures that are used for Hermes will certainly inform what needs to be done in the power reactor, so that it provides basically a knowledge base so that you have confidence that your procedures are sort of the right ones. You're starting, you know, up the learning curve, if you will.

MR. PEEBLES: Absolutely. And that's a

great example of where the iterative learning approach come into play.

CHAIRMAN PETTI: Right.

MEMBER REMPE: So, Dave, I had question I wanted to ask. So I ran something through the CP application. I didn't see anything about your plan. I don't know if you call it a capacity or availability for the Hermes reactor. Do you have any idea how much you're going to run it with respect to available time? Are you planning to run it once a week or, you know, an hour a week. Or have you guys thought about that very much yet?

MR. PEEBLES: No, we don't have that detail at this time.

MEMBER REMPE: Because I think that would be important if you're going to demonstrate how, you know, again I'm thinking about what happened with Fort St. Vrain and availability to have it operating a lot.

MR. PEEBLES: Right, appreciate the comment.

MR. PEEBLES: All right, so next I'm going to turn it over to Brandon Haugh, the senior director of Modeling and Simulation.

MR. HAUGH: Hi, thanks. This is Brandon Haugh again. I'm going to introduce the first slide

here where we just introduce the fuel form, and then I'll turn it over to the manager of Core Design, Nader Satvat, to go into a little bit more detail on the design of the core and tools.

So as some of us have seen before, we're using a pebble fuel form. This pebble fuel form has three regions. It's got a lower density graphite in the center of it, that's to maintain the buoyancy of the fuel in the side coolant. It's got a fuel region that's on the outside of that low dense region, and then it's got an outer fuel-free shell designed to protect the fuel region and prevent salt ingress.

That fuel region contains particles that are based on the AGR program for qualification, very similar specifications. For sizing, you can see on this slide that that pebble is roughly the size of a ping pong ball, about four centimeters in diameter.

Core design then is a pebble-backed concept where the pebbles circulate from the bottom to the top of the core, since they're buoyant in Flibe. And then that core consists of a mixture of graphite moderator pebbles and fuel pebbles for optimum moderation.

MR. SATVAT: This is Nader Satvat, manager of Core Design. The specifics of the design of the

core of Hermes are listed in the table on Slide 10. The power of the reactor is 35 megawatts thermal. The fuel cycle of the core is 190 days average residence time, about four to six passes. This is not fully determined, but that's the range of pass for a pebble.

The discharge burn-up of the reactor is six to eight percent FIMA. The safety parameters of the core, overall negative temperature reactivity coefficients, and also negative fuel and moderator temperature reactivity coefficients, also the void and coolant temperature coefficients are negative.

The methods were calculations for using high fidelity methods, such as Monte Carlo, and also internally developed tool, KPACS, for sharpening of the core. There is a slide about methodology here in a few slides. I'll touch in this a little bit with more detail.

(Simultaneous speaking.)

MR. SATVAT: The power per pebble --

MR. HAUGH: Is there a question?

CHAIRMAN PETTI: I said yeah, I had a quick question. Enrichment, are you going up to the LEU limit even though the burn-up's only six to eight percent?

MR. SATVAT: We're using the upper limit

of HELU.

CHAIRMAN PETTI: Yes, okay. So it's relatively over-enriched relative to burn-up?

MR. SATVAT: Yes.

CHAIRMAN PETTI: Yes, I got you. Thanks.

MR. HAUGH: Thank you.

MR. SATVAT: The power per pebble is about 1,000 watt per pebble. That is to say within the qualification limit of TRISO. The pebble figure factor in this core is approximately two. The coolant is Flibe, enriched with Lithium-7. And the level of impurity in the Flibe is also a parameter that needs to be adjusted, in part, to heavy metal at a ratio to get the desired temperature reactivity coefficient for Flibe.

MEMBER MARCH-LEUBA: Yes, this is Jose March-Leuba. Obviously the Hermes core is tenth of the volume of the real reactor. How do you get to critical? What parameters do you change to obtain criticality?

MR. SATVAT: Thank you for the question. There are two approaches that we're considering. One of them is similar to how HDR10 went to criticality. So we call that a layered approach. So slowly inserting -- so the core at the beginning is filled

with graphite pebbles and slowly inserted fuel and graphite with a desired ratio until we get to a critical weight (phonetic). So that's one approach.

The other approach is called mixed approach. And step by step, we are going to increase the ratio of fuel to graphite and natural uranium pebbles until we get to criticality.

In both approaches, the prediction of next step is very similar to how all conventional reactors are done with one-over-M approach to get there safely.

MEMBER MARCH-LEUBA: But doing some correcting, are you planning to change the configuration of the core, that you run into graphite dramatic concentrations?

(Simultaneous speaking.)

MR. SATVAT: -- ratio of the pebbles, yes.

MEMBER MARCH-LEUBA: So if you are going to do it experimentally with one of them, it's going to take you a couple of years to do the startup.

MR. SATVAT: The layered approach is not going to be time consuming as opposed to the mixed bed approach. Currently PHSS is capable to re-circulate the whole core in less than 72 hours. And currently the calculations we have done, it takes about six to ten steps to get to criticality.

So you are correct. The mix of that is going to be taking time. But it also allows us to do some tests on the way they're specifically checking on the condition of the fuel as they circulate through the core.

MEMBER MARCH-LEUBA: How fast can you re-circulate the whole core? I'm concerned about the homogeny to your core, that you start filling it up from the bottom, and you risk criticality. And now you still have a non-critical pump that you are going to go super critical when you put more. You see what I mean? You have to swap it and make it homogeneous.

MR. SATVAT: That is a very good point. At each step of the mixed bed approach, the whole control rod system is fully inserted. So the prediction for next step, the next step starts with all the rods in and slowly withdraw. The predictions are calculated based on fully withdrawn control system. So if in any case that next step we're mismatching and additional, extra pebbles, the control reactivity will basically compensate that.

As far as answering your question for PHSS, I'll hand it over to Nico to respond to that question.

MR. ZWEIBAUM: Yes. Well, so hi, this is

Nico Zweibaum. I'm the director or Salt Systems Design which encompasses the pebble handling and storage system.

But as Nader was just mentioning, we are currently dimensioning the pebble re-circulation system to be able to re-circulate the full core in about 72 hours. These parameters may be adjusted based on what comes out of core design optimization and alterations, but the reality is that the hardware is pretty flexible to adapt to what the needs might be on the core physics side.

MEMBER MARCH-LEUBA: Yes. And I'm sure you've thought about this. And I certainly would like to see all the details. But when you do the re-circulation, do you remove uranium pellets and replace it with a carbon pellet? Or how do you ensure homogeneity if you are doing it on the fly? I'm sure you thought about it, but I want to see the details.

MR. SATVAT: Sure. As far as the mechanical design, you'll see an animation that will give you a better sense of how we are sorting pebbles, and extracting them, and reinserting them. And then we may follow-up with more specifics after that.

MEMBER MARCH-LEUBA: Right. And the other thing is will you use the Hermes reactor, which a

fantastic thing that we're doing it, to calculate all those relativity coefficients. But there will be -- how do you measure on, you run into carbon ratio that is different than in the real reactor. So we'll have to extrapolate, based on calculations, to what 140 megawatt electric will do, right?

MR. ZWEIBAUM: Yes, that's correct.

MEMBER MARCH-LEUBA: Okay. Thank you.

MEMBER REMPE: I have a question about your rods that are in the core that are designed to go in the core. As I recall, the THTR had some damaged pebbles from that. And why are you sure you're not going to have the same problem? Because actually that led to some unavailability with the THTR.

MR. SATVAT: This is Nader Satvat, manager of Core Design. I'm going to hand it to Chad Nixon, the responsible engineer for the testing around that component.

MR. NIXON: Hi, this is Chad Nixon. We've done testing already with shutdown elements. And one of the main things here is that the elements are inserting into a bed that the pebbles are positively buoyant. There's much less force required to insert into our pebble bed since the pebbles can't depress down into the core at the point the elements are

inserted.

MEMBER REMPE: That's good. And the, quote, pebbles in your test were actually the same material that will be in the core.

MR. NIXON: The preliminary testing we've done is scaled testing with plastic polypropylene pebbles.

MEMBER REMPE: They are floating in some sort of fluid, I guess?

MR. NIXON: In water, yes.

MEMBER REMPE: Is some write-up about that available for us to see? Again, I only looked at the CP and a couple of the topical reports. Because I didn't see anything about we've done testing, and we have confidence that this is going to be okay.

MR. NIXON: No, we're not including that as part of the construction permit application.

MEMBER REMPE: Okay, thank you.

MEMBER MARCH-LEUBA: Well, I'm reading ahead, and I read this slide. And I'm looking at the Slide 11. The shutdown margin we're shooting for is k-effective of .99, which I assume is a non-proprietary number. I realize this is with one full rollout. But that's only two rollouts from critical. We will not make any mistakes while we're rolling all

those graphite pellets. And it doesn't sound like too much margin to me. But, just a comment.

MR. SATVAT: This is Nader Satvat. Thank you, Dr. Jose March-Leuba. That's a very good point that you're bringing up. If you look at our application, there is a lot of margin in our control system. That is just the bare minimum. But, on top of that, we're actually recognizing it's first-of-a-kind reactor, we do have, I believe, about 4,500 PCM extra margin in our control rod system.

MEMBER MARCH-LEUBA: So then those 4,000 PPM is control rod systems you don't credit, but exists.

MR. SATVAT: It does exist, precisely.

MEMBER MARCH-LEUBA: Okay, thank you.

MR. SATVAT: Yes. And just to add one more point to previous question, in this reactor, the fusion length is about to eight to ten diameter. There's some level of biasing in the bed, not complete homogeneity. It's not going to change parameters in the core. However, recognizing that mixing the bed during operation is a parameter that we need to take into account, we do have an uncertainty analysis which looks into perturbing or biasing carbon to have a metal atom ratio across the core and observing the

impact on safety parameters.

Now I'm going to go to Slide 11. The other reactivity control and shutdown system is, as demonstrated on the right picture, has three shutdown elements going directly to the bed, and four reactivity control systems in the reflector. For that to run, the director of reactor systems will go into more detail about the release mechanisms and the diversity right after this session.

The shutdown margin compensates power defect, xenon decay, operational excess reactivity, and depletion of the rods. As was just discussed, the shutdown margin takes into account a single most reactive rod failure and 1,000 PCM to k.1.

The sources of operational excess reactivity, core composition is one of them. And it's determined for different core states to compensate change, for changing power levels, or manage other transients. The method, we do have a high validity method to calculate the power defect which combines Monte Carlo and Kairos media using Star-CCM as the tool.

Other notes. Drive mechanism sets limit on withdrawal rate, which is the rate of insertion of reactivity. And, also, KP-FHR has a strong prompt

effect to reduce regular use of the RCS.

The next slide is the core design methodology. There are three boxes here. The box on the left, which is the green box, light green is the safety tools. And as discussed earlier, due to lack of operational data for FHRs, pebble bed FHRs, we are relying on high fidelity methods, SERPENT as other Monte Carlo engine reactor physics calculations, and Star CCM for our pedal. The core, core is media approximation.

Also Star-CCM is used for discrete element modeling which determines the flow of pebbles in the core and their distribution of resident's time which is an input to KPAX. KPAX is an internally developed tool to do field cycle analysis for pebble, but very similar to VSOP but higher fidelity. KPATH is another internally developed tool which connects a couple, SERPENT and Star-CCM.

We do generate -- we do process of our own ACE (phonetic) libraries for input to AXIOM, PSAB, and SERPENT. That process is a part of our software quality domain.

We also do have a light red box called Support Tools. They're not used in our safety analysis domain, but they are used for design purposes

or understanding the transient behavior of the reactor. And KP-AGREE is a familiar code to threat advanced gas for reactor evaluation. The KP version of that is for Flibe system which is a time-dependent, thermal-hydraulic, and kinetics tool couple.

And the next slide, there are some representative information about the behavior of the core with steady state data. Just for some understanding of these numbers in Hermes' core, on the left side we do have thermal plugs and ASP plugs. As it can be seen, the thermal plugs peak in the reflector, the reflector agent. At the middle there are two temperatures for Flibe, and also the surface temperature, and the distribution of that.

And on the right side is the power distribution in the core, power density distribution in the core, and also in the de-fueling region above the core. And with that --

CHAIRMAN PETTI: I just have a question, given the small size of Hermes, I assume it's relatively leaky in terms of, you know, neutrons are outside the vessel. Is that --

MR. SATVAT: Yes, precisely. That's accurate.

CHAIRMAN PETTI: Yes. So the shielding is

going to have to take care of that.

MR. SATVAT: That kind of hits, that's a correct point. In the -- Hermes says design, reflector does a relatively good job for reducing that, but still, you're right, we're taking that into account for sure.

CHAIRMAN PETTI: Okay. So before we move on to Oded, I just wanted to point out that we're slightly behind schedule on our planned time allotment for each slide presentation. So I just wanted to check with Weidong and make sure that's that okay if we start eating into the closed session time, maybe take some of that back.

MR. WANG: I think up to Dave. Dave, how do you think?

CHAIRMAN PETTI: No, let's just keep going. Okay, Obed?

MEMBER MARCH-LEUBA: Yes, the time -- this is Jose. The time estrangement is always the fault of the members, and you can blame us for that. Keep going.

(Laughter.)

CHAIRMAN PETTI: Got it.

MR. DORON: Okay, can you hear me okay?

CHAIRMAN PETTI: Yes.

MR. DORON: Okay. I'm Oded Doron, Director of Reactor System Design. I'll be mainly focusing today on high level overview of the vessel internals and reactivity control and shutdown system. Again, the content here is relatively high level. We've tried to pull things directly from the PSAR whenever possible so just give you a summary.

So a simple diagram here where you can see on the left a vessel, lower head, coolant inlet nozzles, and the vessel top head. Onto the right we get into the internals.

The core structure is formed by graphite blocks. Starting at the bottom we have a reflector support structure that initially the reflector blocks will sit on until the Flibe enters the system.

And then the blocks will -- they're buoyant, so they will float, fueling chute, lower fueling chute, the active core region, graphite reflector, the core barrel which is concentric with the vessel, downcomer region which is formed between the core barrel and the vessel, upper plenum regions, the fueling chute, and the flow diode which is utilized for a natural circulation shutdown event.

CHAIRMAN PETTI: Just a question. Go back. You probably may not be there, but you probably

know, pebble beds, the stress on the support plate, you've got a lot of the pebbles to come through. You have less graphite there. Have you guys gotten to the stress analysis stage to see that you don't have any problems exceeding limits on the support plate?

MR. DORON: Yes. So the support plate will only --- we have to hold the weight of the graphite structure until the Flibe enters the system, and then the Flibe will float. And so the support plate will be essentially stress free, the lower head in general. It will have to support the weight of the Flibe but not of the pebbles or the graphite. You'll have to remember that they're buoyant.

CHAIRMAN PETTI: Yes, okay.

MR. DORON: Okay. And the weight of the pebbles that are inserted, I think Nico will be touching a little bit on that earlier. But they do not go through that support plate. And yes, we have started conducting stress analysis on the graphite.

CHAIRMAN PETTI: Okay. Thanks.

MEMBER REMPE: How big is -- what was the diameter and height of the vessel? I didn't see it in the PSAR.

MR. DORON: I don't believe it was provided.

MEMBER REMPE: It doesn't have to be exactly. I mean, is it three feet or --

MR. DORON: Eight-ish feet in diameter, you know, let's say 12 to 16 in height, something like that.

MEMBER REMPE: Thanks.

MR. DORON: Yes.

MEMBER BROWN: Can you stay on that? This is Charlie Brown. I'm now back to the Slide 15 again. I don't understand pebble bed reactors. You've gone through this before, and I think I've forgotten. All of the pebbles are inside the thing you called the active core. They come up from the bottom, they go out the top. Is that correct?

MR. DORON: Yes, sir. That is correct.

MEMBER BROWN: And the cooling means the Flibe, is that outside, or does that get mixed with the pebbles as well?

MR. DORON: The Flibe is everywhere.

MEMBER BROWN: So it's within the vessel as well as external to the vessel?

MR. DORON: Not outside of the vessel, no. Inside the vessel, within the vessel structure.

MEMBER BROWN: Well, you said the graphite reflector is outside the vessel --

MR. DORON: Right.

MEMBER BROWN: -- and when the Flibe comes in, that the graphite floats.

MR. DORON: If I said that, I misspoke. So let me say it a little different maybe. The graphite is in the vessel. Initially when we load, we load without Flibe. We load dry.

MEMBER BROWN: Well, hold it. Maybe I'm calling the vessel the wrong thing.

MR. DORON: Okay.

MEMBER BROWN: I'm talking about that little tube in the center.

MR. DORON: Oh. No, sir. That is the core region that is formed by the graphite structure. So you have a lower plenum or fuel chute and then upper plenum and de-fueling chute.

MEMBER BROWN: Oh, okay. So it's not like there's a container that the pebbles sit in. They --

MR. DORON: No.

MEMBER BROWN: -- come up through an annulus within the graphite reflector.

MEMBER MARCH-LEUBA: Charlie, this is Jose. Maybe you can show us the Slide 16, show us the flow of the coolant.

MR. DORON: Yes, that can help.

MEMBER MARCH-LEUBA: And we'll understand better.

MR. DORON: Yes.

MEMBER BROWN: See, I looked at that one to see if I could --- I'm sorry, I looked at that one. I was lost there too, so I apologize. Go ahead to 16 if that'll help.

MR. DORON: That might help. And let me, yes, so let me make a comment here and maybe this comment will help you, make it a little clearer, is that the internal structure is formed by the graphite structure, okay.

MEMBER BROWN: Okay, but this little barrel in the middle --

MR. DORON: That is formed by the graphite.

MEMBER BROWN: Okay. And that's where the pebbles are contained as they flow in --

MR. DORON: Correct.

MEMBER BROWN: -- and then up through. And that's the blue stuff in the left-hand side? Or is that the coolant flow path?

MR. DORON: That is the coolant flow. This is all coolant here. This is not pebbles.

MEMBER BROWN: Okay. But the pebbles and

coolant mix, right?

MR. DORON: Yes. So again, Nico will touch on that later in his presentation. So we have pebble insertion lines. And so pebbles are inserted through, and essentially through the graphite structure, if you could think of it like that. And then they enter through the lower fueling region. And they float their merry way up through the core.

MEMBER BROWN: Does the graphite or the Flibe go out along with the pebbles?

MR. DORON: The Flibe out of the free surface at the top, at the top of the vessel.

MEMBER BROWN: So somehow the Flibe and the pebbles get separated?

MR. DORON: Correct. Nico will go into that.

MEMBER BROWN: Okay, all right. I'll stop then. I won't slow this process down.

MR. DORON: Okay.

MEMBER BROWN: I'm sorry, I just don't know pebble bed reactors.

MR. DORON: These are good questions. These are good questions.

MEMBER REMPE: This is Joy. And I am thinking about what I saw in the PSAR about the vessel

being fabricated and tested to have an extremely low probability of leakage. And of course, I know you're only talking the portion the vessel that holds the coolant. Because I know you want to keep the coolant above the core to make sure that it provides a fission product release barrier. And I'm curious if you have a specification for what an allowable leakage is. Because everything leaks a little bit in life, it seems like.

And secondly, how much above the core is -- I never saw something like would give me an idea whether it has to be an inch above the core, a millimeter above the core, or a foot above the core. Can you give me an idea of what you guys are thinking about? Because I, again, didn't see it in the PSAR.

MR. DORON: Yes. Let me take those one a time. So first of all, we are not assuming that the vessel will leak Flibe. That is not an assumption we're going with. So --

MEMBER REMPE: That's zero leakage, they can't have any sort of leakage at all.

MR. DORON: I mean, if you think of it as a -- it is the vessel that is containing the, you know, all of the structure and all of the Flibe. And so if I were to have leakage, it would be some kind of

a failure. It's not, I mean ---

(Simultaneous speaking.)

MEMBER REMPE: Let me interrupt you and put it in a different way. I used to do leak testing on sensors. And so even when we did leak testing, there was a little bit of leakage. And that was considered acceptable. And you're saying you're going to have a perfect system that just isn't going to leak at all.

MR. PEBBLES: So this is Drew Pebbles again. That is correct, that there is no leakage that's going to be allowed from the vessel. And all the penetrations are above the free surface of the Flibe.

MEMBER SUNSERI: I would add that's not unlike a PWR that has zero pressure boundary leakage as a criterion. And if you do get a leak, you have to shut down.

MEMBER REMPE: When we used to do it, it was like something like ten to the minus ten or something --

MEMBER SUNSERI: No, I know, but there's controlled leakage, there's pressure boundary leakage.

MEMBER REMPE: Having the penetrations above helps, I can get that, and your wielding below.

So I get that. But go ahead and answer the other questions, please, about how much above the core.

MR. DORON: I'm not sure if we defined that in the PSAR, whether or not, but the fuel will remain covered. So the coolant cannot drain lower than covering the fuel.

MEMBER REMPE: Okay. So I think that's going to be important to understand, because you're going to have to have instrumentation to understand when to get worried about that it's getting too close to the top of the core. And I'm guessing you don't want it just right level with the core.

But it'll be interesting as we evaluate the instrumentation to make sure that there is enough above the core that the sensors give signals to the operators saying we've got a problem, and we need to do something.

MR. DORON: Yes. And when Anthony, our director of Instrumentation, Control and Electrical speaks later, maybe he could touch on that just a little bit. But you're correct.

CHAIRMAN PETTI: So but just to ask the question a slightly different way, how far from the top plate is the Flibe level?

MR. DORON: I don't believe we define that

--

CHAIRMAN PETTI: Okay.

MR. DORON: -- in the PSAR either. But, I mean, you know, you could throw a number out. I mean, some several inches, something like that if you like it, during normal operations.

CHAIRMAN PETTI: Yes. Only if you want to stay below those penetrations, you know, you've been, yes. Okay.

MR. DORON: Correct. Good questions. Okay. Let me jump into the flow here, okay. So if you look on the left, normal operation coolant flow path, I have the flow entering through the inlet nozzles. And you'll recall I mentioned that the core barrel is concentric with the vessel and the gap between the core barrel and the vessel forms our downcomer.

So I have a cold inlet coming through the nozzle, through the downcomer, all the way down, coming around and up through the core region that's formed by the graphite, up through the upper plenum and out the top.

So during natural circulation, I have similar flow path, except I don't have my pump anymore. And so it's not coming through the inlet,

and it's not coming out the top.

Instead I have the flow coming naturally up through the core, heating up and then making its way through the flow diode, and then into the downcomer region where the heat is pulled out with our DHRS, or decay heat removal system which is on the outside of the vessel. And Nico will touch on that later. It cools down through the downcomer, and then repeats the process.

MEMBER MARCH-LEUBA: So let me see if I understand. This is Jose. On the left, see on the left when you're pumping, when you have a flow diode, the red coolant goes out of the vessel to the DHRS, correct, and then comes back? I don't see a red arrow coming out.

MR. DORON: Right. So the DHRS does not take the actual coolant. The DHRS removes the heat from the vessel wall.

MEMBER MARCH-LEUBA: That is for natural circulation.

MR. DORON: Yes.

MEMBER MARCH-LEUBA: I said the inside as decay heat. I meant the normal operation of --

MR. DORON: Normal operation, the PHTS, yes, the coolant comes out the top, out of the top

around and back through the inlet.

MEMBER MARCH-LEUBA: Okay. That's ---

MEMBER BROWN: And so going back, it's only back pressure from the inlet that keeps the hot stuff from going out through the diode.

MR. DORON: Yes, sir, that's correct.

MEMBER BROWN: In the left-hand one.

MR. DORON: Correct.

MEMBER BROWN: And is there something up at the top where the red arrows, the red stuff goes out through one of the pipes up at the top?

MR. DORON: Yes, there's a pump. But we're not showing the pump here in this diagram --

MEMBER BROWN: Oh, okay. And then it goes back, it goes around and gets cooled?

MR. DORON: Correct.

MEMBER BROWN: Is that Flibe only, or is that -- are there pebbles mixed in with that as well?

MR. DORON: Flibe only, hopefully. Yes, Flibe only.

MEMBER BROWN: So even though the Flibe and the pebbles are mixed down in the core region --

MR. DORON: Correct.

MEMBER BROWN: They just --

MR. DORON: They cannot enter the hot

plenum.

MEMBER BROWN: That little triangle at the top?

MR. DORON: Okay, so let's go back a slide, Drew.

MEMBER BROWN: How does he get separated?

MR. DORON: So do you see where it says upper plenum there?

MEMBER BROWN: Yes.

MR. DORON: Yes, so flow makes it into the upper plenum, but pebbles do not. So graphite is extremely machinable, extremely machinable. We can make almost any shape to our heart's content within reason, obviously. But we have it designed in such a way that flow enters the upper plenum, but pebbles do not.

MEMBER BROWN: Is it based on physical size?

MR. DORON: Correct.

MEMBER BROWN: And, oh, geez. Okay.

MEMBER MARCH-LEUBA: So can I say that this reactor is going to be 3D printed?

(Laughter.)

MR. DORON: You could say that, but it's not going to be, no. It going to be --

MEMBER MARCH-LEUBA: Three-D hulls.

(Simultaneous speaking.)

MR. DORON: I mean, it's worthwhile to say that, you know, we're machining a full graphite structure currently for our engineering test unit that, you know, drew had in his model there. So this is not conjecture here. We're actually doing this.

And what I will tell you here is that -- and if there's more I think I'd prefer to take it to the closed session. The graphite is extremely machinable, very, very machinable to very, very high tolerances.

MEMBER BROWN: But how does the Flibe and the pebbles get differentiated? I mean, is there something --

MR. DORON: The pebbles can't --- there's coolant paths that restrict flow, that don't allow pebbles into there.

MEMBER BROWN: But the pebbles can't block it?

MR. DORON: Well, correct. Because the pebbles are continuously moving.

MEMBER BROWN: Well, so is the Flibe.

MR. DORON: So is the Flibe.

(Laughter.)

MR. DORON: Yes. Yes, sir. Yes. Again, I think -- I'm hopeful, maybe we can circle back after Nico's presentation when you see a little bit --

MEMBER BROWN: All right.

MR. DORON: Let's circle back after that and see if this, in combination with his presentation, help answer, help shed some light on your questions.

MEMBER BROWN: Okay.

DR. BLEY: This is Dennis Bley. Is there any chance, for the closed session, you guys have some movies that would let people understand this better?

MR. DORON: I'll leave that to Drew. We have some PHSS movies that we've done. I don't know --

MR. PEBBLES: Yes, we have some animation for the PHSS presentation. So we can circle back after that and see if it helps clear up some ---

MR. DORON: Yes.

MR. PEBBLES: We don't have any backup slides though, only what we submitted on the topic.

MR. DORON: Okay, shall we continue? All right. This is the head layout. So, I mean, I can go through every one or not here. But the big items, you know, the pump is on the head. You mentioned the coolant level sensors. You could see the allocated

space for those.

The shutdown elements that are in the core, there's three of those. They're indicated by the red dash circles there, and then the 4X for control elements by the yellow dash circles there. And then you could see we currently have two means of pebble insertion, material sampling boards, reactor thermocouples, a location for a neutron source.

MEMBER BROWN: So you have two types of ---

MR. DORON: Sorry.

MEMBER BROWN: You said this earlier. So you've got two types of reactor. Okay, one of them's a shutdown, okay, so one of them's shutdown, the other one's control elements.

MR. DORON: Correct. And the next slides are going to be discussing those.

MEMBER BROWN: And if one -- and you all are doing, I think you mentioned this earlier, but if one reactor shutdown accident element doesn't operate, that's a pretty thin margin. I think Dave or somebody made a comment about that.

MR. DORON: Yes. But we are allowing for that.

MEMBER BROWN: And where are your sensors,

your neutron sensors? Those are the purple things?

MR. DORON: Yes, source range neutron detectors.

MEMBER BROWN: What about power range or in between, whatever your ranges are?

MR. DORON: I can let -- Nader, do you mind speaking to the power detectors quickly?

MR. SATVAT: Sure. This is Nader Satvat. The power range detectors are in the cavity, in the bio-shield structure.

MEMBER BROWN: In the what structure?

MR. SATVAT: They're outside of the reactor vessel.

MEMBER BROWN: Oh, so they're external to the vessel. These are in core -- the source range are in core, the other ones are ex-core.

MR. SATVAT: Yes.

MR. DORON: Ex vessel, rather.

MEMBER BROWN: Ex vessel, that's fine. I meant ex vessel.

MR. DORON: Okay.

MEMBER BROWN: Okay. Well, not okay, I'm just saying I got you.

(Laughter.)

MEMBER MARCH-LEUBA: This question might

not relate certainly in your presentation, but I notice there are three detectors. I'm thinking I&C. Are we going to have two out of three detection systems?

MEMBER BROWN: I looked at that, and there's no definition of what there's going to be. It's just a box.

(Laughter.)

MEMBER MARCH-LEUBA: Well, if you have only three detectors you're going to have close to four. So think about it. We'll need to know.

MR. DORON: Okay. And maybe Anthony can take that later.

(Simultaneous speaking.)

MEMBER REMPE: What is the reserve instrumentation? What are you going to put in there?

MR. DORON: You know, whatever it is that we think is appropriate at the time. This is a first of a kind facility.

MEMBER REMPE: So I have seen thermal couples listed, and I've seen the level detectors.

MR. DORON: Yes.

MEMBER REMPE: I'm kind of wondering what else you're going to put in.

MR. DORON: There's lots of things we're

discussing. We're leaving it available.

MEMBER REMPE: Okay.

MEMBER BROWN: What's a neutron source?

MR. DORON: I don't know if we specifically discussed which one is our neutron source in PSAR.

MEMBER BROWN: I thought uranium fissioned and produced its own neutrons.

MR. DORON: This is just for startup --

(Simultaneous speaking.)

MR. DORON: -- the neutrons are the startup source.

(Simultaneous speaking.)

MEMBER BROWN: You need an external source in order to start up the reactor?

MEMBER KIRCHNER: Charlie, you always put a external source in the core to start up.

CHAIRMAN PETTI: Yes. It's either PuBe or americium-beryllium, usually.

MEMBER KIRCHNER: Yes, americium-beryllium is a common one. But almost all reactors have that. So you have the signal when you begin startup. And this goes towards those earlier questions.

MEMBER BROWN: I will not make any comments on that.

(Laughter.)

MEMBER REMPE: So have you thought what type of coolant level of the sensor you're going to use yet?

MR. DORON: We have thought about it. And again, I don't know if Anthony's going to go into that level of detail.

MEMBER REMPE: Well, it's actually not in the PSAR too. I mean, you mentioned thermal couple, I don't know what kind of thermal couple, but the other detectors are pretty much undefined. And I assume it's not going to change before the PSAR is finalized.

MR. DORON: Drew, do you want to let Anthony go here, or do you want to take a note to discuss those later?

MR. PEBBLES: Let's get to Anthony's presentation, just in the interest of time. And he can speak to the level of detail that we have in the PSAR for the InP system.

MR. DORON: Okay. Okay, so Hermes Reactivity Control and Shutdown System, again, this is relatively high level. It's from the PSAR. So the 4X core in the reflector, and 3B core shutdown elements, so the ones in the reflector are control elements, the

ones in the bed are shutdown elements.

The drive mechanism is a motor-driven sheave. It's to position the elements. And release mechanism is those two release, essential release mechanisms. One is an electromagnetic clutch, and the second is a motor isolation.

If you look at the little diagram there on the right, we have the elements, a counter weight, the wire rope. There's a housing there, connector, the elements to the wire rope, the sheave, the clutch, and the motor. That is all.

On the left we have the control element, and on the right we have the shutdown element. So again, the control element enters a dedicated path in the reflector structure, the element connector there that connects it to the wire, we have the cap, the element connection plates, control element segments.

The control element is a segmented annular design. It's got individual capsules, argon filled. The absorber is B4C, the cladding is stainless, 316H, a little diagram there showing what a cross section of the control element might look like.

The shutdown element is cruciform. It's got, again, the connector on the top plate there, it's cruciform design, inner cladding, it contains the

absorber. It's argon filled. And the absorber is B4C. And the cladding is stainless steel 316H.

That is all. I appreciate all the questions. Thank you very much for your time.

MEMBER REMPE: So you don't have to answer it now but later. I know I saw in the PSAR the comment about the B4C melting temperature was more than 1,000 degrees C above the operating temperatures. But I didn't see anything about liquefaction temperatures with B4C and stainless steel.

And are you considering that too? Because it seems like that's a lower temperature than the melting temperature. It's still probably not a problem, but you might want to think about it.

MR. DORON: Okay. We can take that one as a note in the interest of time.

MEMBER REMPE: The other one to think about, that I was wondering about was reading about this. The PSAR dismisses any concern about combustible gas generation. And I get that you may not be so concerned about hydrogen, but what about, is there no concern about any sort of carbon-related combustible gas generation?

MR. DORON: You know, a good question. I would go to our salt chemistry team to answer that.

I don't know, I'm not ---

(Simultaneous speaking.)

MEMBER REMPE: And even if it's not today, it's just something to think about for future discussions.

MR. DORON: Okay.

MR. PEBBLES: We'll take that back. Thank you.

MR. DORON: Yes, appreciate that.

MR. ZWEIBAUM: All right, good afternoon. My name is Nico Zweibaum. I'm the director of Salt Systems Design at Kairos. I'm going to be talking about heat transport as well as pebble handling and storage system in Hermes. So my mission here is for everyone to understand where the Flibe goes, where the pebbles go, and everything around that.

So starting with our primary heat transport system, or PHTS in short, that system in Hermes is responsible for transporting the heat from the reactor to the ultimate heat sync, which is air, during power operation and during normal shutdown.

That system is carrying Flibe around. It operates near atmospheric pressure. It does not provide a safety-related heat removal function. The safety-related heat removal system is our decay heat

removal system which I'll be talking about next.

One safety-related function of the PHTS though is a hot lag anti-siphon feature. And that is performed by the geometry of our primary salt pump. That pump, which we alluded to but did not show in Oded's talk, it's sitting on the vessel head. The reactor vessel has this upper head, and the pump connects to it.

And it has a downward facing inlet. That's here the Flibe is going through the pump and out to the hot lag. When the level drops, that pump essentially deprimes, and this what's providing that anti-siphon feature. So we're not draining coolant outside of the vessel, especially not below the normal operating levels to keep the fuel covered.

A number of additional functions for that PHTS, it contains the reactor coolants and directs the flow between the reactor vessel and the heat rejection sub-system. It is equipped to manage thermal transients, maintain overall thermal balance that's occurring as part of normal operations.

Since our coolant has a relatively high freezing, melting temperature, it is equipped with features to ensure that we maintain acceptable minimum temperatures through makeup heating as necessary

during operations. It is able to drain to reduce for acidic heat losses when we run into overcooling transients. And it does provide for in-service inspection, maintenance, and replacement activity.

MEMBER BROWN: How hot do you have -- how high do have to keep the temperature for the fluoride salt to keep it from solidifying?

MR. ZWEIBAUM: So the freezing temperature of our Flibe is, I believe, around 460 degrees Celsius. You'll see on Slide 23 our normal operating temperatures, but the minimum nominal temperature during operations is 550 C, so almost 100 C above freezing.

MEMBER BROWN: So you have to keep it 100 degrees C above freezing at all times with another system?

MR. ZWEIBAUM: You don't have to keep it that high. That's the nominal temperature. You do want to maintain a healthy margin above freezing though, and this is what that makeup heating system does, so maintaining it above the freezing temperature of 460 C.

MEMBER BROWN: And how do you maintain that uniformly throughout the system.

MR. ZWEIBAUM: We do have a thermal

management system that consists of a combination of heaters and insulation. And we'll have a number of demonstrations along the way to ensure that we know how to handle Flibe and keep it molten in those systems.

MEMBER BROWN: Is it stationary during that period of time? Or is it still being pumped?

MR. ZWEIBAUM: It will be pumped at most times except during a number of transients like system blackout, for instance, where you would lose your pump. And this is when we would get into the more safety-related decay heat removal which I'll be talking about in a moment.

MEMBER BROWN: Okay. Thank you.

CHAIRMAN PETTI: Just a question. This whole issue of the draining, and preventing freezing, does that make that part of a system safety grade or not?

MR. ZWEIBAUM: No. The safety case is really around keeping the vessel and the fuel intact. And so it's really about maintaining low enough temperatures to not compromise the integrity of the vessel service.

CHAIRMAN PETTI: But if you were to freeze the coolant, that would not be a good day, right?

MR. ZWEIBAUM: Hold on a second.

(Simultaneous speaking.)

CHAIRMAN PETTI: Would that lead to an event you need to ---

MR. ZWEIBAUM: That is more of an investment protection feature and less of a safety concern. It is something that we'll want to maintain for our own, I mean, to maintain the plan. But that is not in the safety space.

MEMBER KIRCHNER: Dave, this is Walt. Could we ask anyone, everyone on the line who is not a speaker to mute their microphones. We've got background noise. Someone's having lunch somewhere out there.

MEMBER REMPE: This is Joy. And I think from the way the colors are flashing on the screen this is coming from the conference room which may make it hard to mute yourselves if you're talking, but think about it, okay.

MR. ZWEIBAUM: Yes, I think people are being pretty disciplined here, but we'll keep it in mind.

I'll keep going here, talking through what that PHTS system is made of as far as subsystems or components. We have our reactor coolant, obviously,

which is Flibe, that primary salt pump I mentioned before, which is a variable speed cartridge-style pump that's attached to the vessel head. It's inlet extends downwards through the free surface, and this is how the coolant gets into the pump and out to the PHTS.

We have a heat rejection subsystem. That subsystem provides for heat transfer from the reactor coolant to atmosphere. It consists of a radiator, a heat rejection blower that circulates air across, and associated ducting and thermal managements.

We have our primary loop piping which is what the Flibe is circulating through, and primarily thermal management which, as mentioned earlier, provides non-nuclear heating and insulation as needed for various operations to keep the system at desired temperatures.

On the next slide, this is a table and a figure that, I believe, are actually strictly from the PSAR but giving you a very, very rough sense of how the system is configured on the right. So you do see the reactor vessel in a much less exciting fashion than what Oded was showing.

But you do see the primary salt pump on the upper right of that vessel, through which the

coolant goes out off to the side to the heat rejection radiator. This is where heat transfer occurs to the atmosphere with that heat rejection blower and stack. And the coolant comes back through coal bags back into the reactor vessel.

The thermal duty, as mentioned in the core design portion, is 35 megawatts thermal. We do plan on having a single heat rejection radiator, single hot lag, two coal bags to return into the vessel. The primary loop line size is generally envisioned to be somewhere between 8 and 12 inch nominal pipe size.

The hot lag temperature, if you will, before the coolant gets into that heat rejection radiator, will be somewhere between 600 and 650 Celsius, depending on operational modes. The cold lag temperature is 550 C. Nominal flow rate is 210 kilograms per second, and the design pressure is generally estimated at 525 kilopascals.

So this was for the PHTS which is our non-safety-related heat transport system.

MEMBER BROWN: How many kilopascals did you say?

MR. ZWEIBAUM: Five hundred and twenty-five.

MEMBER BROWN: Do you have that in pounds

per square inch?

MR. ZWEIBAUM: I will in one second, 76 psi.

MEMBER BROWN: Okay, thank you. I came that close to that. All right, I just want to make sure I was right, thank you.

MR. ZWEIBAUM: That is the design pressure though, that is not the anticipated operating pressure.

MEMBER BROWN: How much lower do you anticipate that to be?

MR. ZWEIBAUM: We'd be closer to atmospheric pressure during normal operations.

MEMBER BROWN: Okay. All right, thank you.

MR. HUGHES: This is Joel Hughes. I'm the responsible engineer for the primary heat transfer system. So just maybe one quick clarification. So the carbon gas pressure at the inlet of the pump is quite near atmospheric pressure. But the pump obviously does add some pressure to it. It will be a fair bit below that 525 kilopascal.

But it kind of depends on what your definition is in terms of close to atmospheric pressure. But it certainly adds some pressure at the

outlet there. And that's consumed along the flow path back around the PHTS through the vessel back to the inlet of the pump.

MEMBER BROWN: Okay, thank you.

MEMBER KIRCHNER: Joel, while you're on the line, this is Walt Kirchner, how does the layout break the seal? You've got the pump inlet stuck, not stuck, intentionally positioned under the free surface. Under normal operation, how many inches, or meters, or whatever measurement you use, is that? And what breaks the suction if you have a break in the primary loop?

MR. HUGHES: Excellent question. So I think as Oded mentioned, we'd have to define the exact elevation of the pump inlet. But that downward facing, basically, inlet of the pump would break the suction on the hot leg side. So if we had a break in the primary salt piping, you could pull a siphon, right, and then down as the level of Flibe in the vessel kind of travels downwards, and Flibe is leaving the system. At some point it would break at the inlet of the pump, specifically above the core level. I don't know exactly how many inches above.

MEMBER KIRCHNER: Okay, okay.

MR. HUGHES: And then there's always a

similar --

(Simultaneous speaking.)

MEMBER KIRCHNER: -- above the core?

MR. HUGHES: Yes, that's the idea. And then there would be, like, kind of a similar feature. It might look geometrically different, but functionally similar on the cold leg side as well so that we don't siphon through the downcomer.

MEMBER KIRCHNER: Right, right. Okay, thank you.

MR. HUGHES: Sure.

MR. ZWEIBAUM: So on to our safety-related decay heat removal system, that is our DHRS, you see a diagram of the configuration here on the left. So the purpose of that system is to provide vessel protection during postulated events for which the PHTS we were talking about previously, is unavailable.

How this system works is based on in-vessel natural circulation. Oded mentioned that before as part of what happens inside the vessel. So you see the vessel represented here. But really what this system is about is what's around it.

So it's a water-based ex-vessel system, and the heat transfer modes are via thermal radiation and convection. It is a system that operates through

continuous direct roll-off when the decay loads exceed for acidic losses. That system is actually shut off and isolated from the system when we're operating at no or low power levels. In that case, we can rely on heat removal via for acidic losses only. We're not relying on the roll-off feature.

And the main thing on operation is that this system gets activated when we cross some power threshold. But after that, the system's status has not changed. The state of that system does not change on reactor event initiation. So when that DHRS, which is a passive decay heat removal system, is called upon for decay heat removal from the vessel, the system is already containing water and ready to boil off.

This is kind of a self-regulated mechanism in that the removal rate is directly a function of vessel temperature. Since the primary mode of heat transfer between the vessel and the DHRS is by thermal radiation heat transfer, which is directly dependent on temperature, which is important since the main metrics that we're trying to control are going to be peak temperatures of that vessel surface.

So you can see the configuration on the left with the vessel that is facing those annular thermosiphons. Those are connected to a water storage

tank via some piping and a separator where there is the separation between the liquid water that comes in and steam that comes out after boil off.

The storage tank is vented to atmosphere, and so as the steam comes out, we do have that getting out from the system through that upper penetration --

MEMBER MARCH-LEUBA: I have a couple of questions.

MEMBER BROWN: Yes, that cavity the vessel sits in, is that just air? And therefore you said radiation from the vessel to the rods or whatever, the annular thermosiphon.

MR. ZWEIBAUM: Yes.

MEMBER BROWN: So there's nothing in there. It's just a dead air space, and then depending on radiation and whatever convection flow of the air within that space?

MR. ZWEIBAUM: That is correct.

MEMBER BROWN: So it's not a wrap-around, is what I'm trying to get, where it's in contact with the vessel?

MR. ZWEIBAUM: That is correct.

MEMBER MARCH-LEUBA: So this is Jose. How many thermal cycles are there, I assume when you designed it? I'm sure there is not only two.

MR. ZWEIBAUM: No, there is not only two. There is a bit more detail on the next slide. So maybe we can go through that. And as far as the --

MEMBER MARCH-LEUBA: Hold on, let me ask my second question. Maintaining the inventory in the storage tank up there, is it a safety function?

MEMBER MARCH-LEUBA: No.

MEMBER MARCH-LEUBA: Because if you run out of it, then you have a problem. And if you are constantly operating, you are constantly boiling it off.

MR. ZWEIBAUM: Yes. So the system is sized for that.

MEMBER BROWN: Pardon?

MR. ZWEIBAUM: The system is -- sorry, I don't want to talk over you.

MEMBER MARCH-LEUBA: Yes, you said that the system is sized for it, but things tend to go wrong sometimes. I mean, I would have, at least in the protection system, or certainly in the alarm system, the level is too low.

MR. ZWEIBAUM: Yes. So there would be some level where we would need to shut down. I may -- we have the responsible engineer for the decay heat removal system, Casey Tompkins, on the line. So maybe

he can speak to that specific question.

MEMBER MARCH-LEUBA: No, just so you're thinking about it, then we'll look. When we have details, we'll look into it.

MR. ZWEIBAUM: Okay. That's all right then. We'll keep going.

MEMBER BROWN: The point is you're venting to atmosphere, so you're going to be losing water when you're really hot.

MR. ZWEIBAUM: Yes. That is part of the operations. We are expecting that.

MEMBER BROWN: And therefore you're going to lose water.

MR. ZWEIBAUM: Can we get to the next slide? Maybe that will get into a bit more detail here that might explain some of this.

So maybe, speaking to the point that was just made, before I go through the contents of the slide, but you can see that that storage tank is actually connected to a feed water system.

So during normal operation, you'll constantly be replenishing that storage tank. However, you're not relying on that feed water when the system is called upon during a transient. So there will be a normal operating mode where you do

have feed water and, I guess, a transient mode or a postulated event mode where you would not have that feed water. And you would be boiling off your inventory that's in there.

MEMBER MARCH-LEUBA: And are you planning to size it for the conventional 72 hours, or those 30 days or --

MR. ZWEIBAUM: Yes.

MEMBER MARCH-LEUBA: It was an either/or.

MR. ZWEIBAUM: Seventy-two, sorry. I started replying after the first half of your sentence.

MEMBER MARCH-LEUBA: Okay, thank you.

MEMBER HALNON: Yes, this is Greg Halnon. Is it just one storage tank, or do you have two, three, just one?

MR. ZWEIBAUM: There are four. So let me go through the contents of this slide. The first point, but obviously by now this is clear, is that that DHRS is independent from the primary coolant. It's a water-based system, so it's isolated from the Flibe system.

One other thing that I mentioned earlier is there is no change of state on setup postulated events. So that system is always on when we cross

some set power level.

But to a question that was asked multiple times, there are four independent cooling loops. And there is a three out of four kind of logic here where we are sizing the system so we can lose one of those four and still be within our envelope.

There is also a dual walls configuration here. So if you look at the symbol, it is contained within a deeper kind of shroud, if you will, so that we can continue to have heat removal in the presence of a water leak within this.

And there is one active component to note here which is an isolation valve between the storage tank and the thimbles, which is closed at no to low power, that gets opened when we cross some threshold power but then remain open.

And so this isolation valve failing in place means that the operating system continues to operate during a postulated event. And then we have a flowed valve inside the separator that passively regulates the flow of water from the storage tank to the thimbles so as not to clog them. But that's during normal operations. But that flow valve not only failed to open so that we don't risk dry out of the thimbles during a postulated event.

MEMBER KIRCHNER: This is Walt Kirchner. Just a further detail. Go to the next slide. It's fine. Yes. Is the system required, probably this would be more tech specs kind of issue, to keep the reactor vessel within its designed thermal limits?

MR. ZWEIBAUM: Yes, we will have tech specs around that.

MEMBER KIRCHNER: Okay, thank you.

MEMBER BROWN: So this, you said there is no change in state relative to postulated events, but it's always on when you cross some power level. So when you startup and you get to some predetermined or calculated power level, then that's system is placed and it's always on configuration.

So it's removing heat during all power range operations --

MR. ZWEIBAUM: Yes.

MEMBER BROWN: -- except if you go below, whatever the number is, and then it goes off again?

MR. ZWEIBAUM: Yes.

MEMBER BROWN: So, some type of sensors tell you that?

MR. ZWEIBAUM: Yes.

MR. PEBBLES: So that's correct. This is Drew Pebbles. I just wanted to make that

clarification. We're going to have a tech spec on DHRS operability, which will likely include level and other things in the tank.

That is for the initial conditions, but it's not for maintaining reactor vessel temperatures during normal operations.

MEMBER BROWN: So you'll also need them, because you've got to feed into it, there has got to be some type of minimum level you allow in the storage tank? I mean, I presume that's part of your overall configurations?

MR. PEBBLES: That's likely what it's going to be for the PSAR level. We're only required to mention the operability tech spec. The specific --

MEMBER BROWN: Yes, that's fine.

MR. PEBBLES: Yes. We'll be providing that with the operating license application.

MEMBER BROWN: Okay. Thank you.

MR. PEBBLES: Yes.

MR. ZWEIBAUM: So the next three slides kind of illustrate, in hopefully more clear ways, what I was trying to describe as far as the three main modes of operation.

So initially, as the core is at low to no power, that isolation valve between the water storage

tank and the thimbles is closed. So the storage tank is full of water, the thimbles are dry. And the only heat that comes out of the vessel is for acidic heat loss, but there is no direct heat transfer to water and the thimbles.

MEMBER BROWN: You said the valve closes, or opens. So it's, somehow it's designed such that it's always going to go open if something fails? Whatever that something is.

MR. ZWEIBAUM: Well, it would fail as is. So in this case we are not relying on having water in the thimbles to extract enough heat. So if there were any failure in, in the current configuration that you're seeing on the slide, then we're not relying on the DHRS for decay heat removal anyways.

MEMBER BROWN: Yes, so the valve would be closed if you go to low power, right?

MR. ZWEIBAUM: Yes.

MEMBER BROWN: But so, something has to make it open.

MR. ZWEIBAUM: Yes.

MEMBER BROWN: If we go up above a certain power.

MR. ZWEIBAUM: Yes, absolutely.

MEMBER BROWN: And what, the failure mode

you're saying is open, but if you're at low power and -- so it's powered to stay shut, and then theoretically if you lose power it opens up? That's just a possible. Is that what you're saying?

MR. TOMPKINS: Hi, this is Casey Tompkins --

MR. ZWEIBAUM: Also -- yes, go ahead.

MR. TOMPKINS: -- responsible engineer for the decay heat removal system. So the valves fail in place. So like if it's open and we lose power or signal to it, it remains open. If it's closed, it remains closed.

MEMBER BROWN: Ah.

MR. TOMPKINS: So if we don't, so if the system is not running it's because we don't need it on a postulated event. So then there is no reason for it to change positions. And vice versa.

MEMBER BROWN: Okay.

MR. TOMPKINS: If it's open, then we need it so it stays open.

MEMBER BROWN: So once it opens it will stay open, if power goes away, and once it's closed it will stay closed if whatever closed it goes away?

MR. TOMPKINS: Correct.

MR. ZWEIBAUM: So this is Nico Zweibaum

again. So, the transition between this slide and the next slide is really what happens as you get above those, this threshold power level.

So this is really normal operation of Hermes. Your core is operating, well, it says high power but really is power above that decline threshold, but is nominal power for instance.

So in this case you got your line between your feedwater and your storage tank that is open. You also open the, or your isolation valve that was between the storage tank and the thimbles is open. And you're continuously flowing water through those thimbles, boiling up and the steam gets vented out to the atmosphere.

And then if you go to the next slide. During the postulated event where you have a reactor trip and you can't rely on your primary heat transport system to extract heat, then you would have this continuous boil-up of the inventory that was in your storage tank.

So we're sizing to not be relying on the feedwater system feeding water into the storage tank, but instead we're boiling up the inventory of water that is in those storage tanks.

MEMBER KIRCHNER: So, might I ask a

question on this? This is Walt Kirchner again. What limits your design here in the three modes of operation, is it the concrete temperature, is it vessel temperature or is it decay heat removal?

MR. ZWEIBAUM: Well, the system is spliced to protect against -- the main metric that we're after is the vessel heat temperature. To avoid failure of that structure.

MEMBER KIRCHNER: Right. Is that during normal operation as well or just under the transient?

MR. ZWEIBAUM: Just under the transient. During normal operations, your main means of heat removal is through the primary heat transport system.

MEMBER KIRCHNER: Of course. But is there, does that keep the temperature, well, I guess the downcomers, the inner wall actually of the vessel. So --

MR. ZWEIBAUM: That's right. Yes. And that's where you have --

MEMBER KIRCHNER: Yes.

MR. ZWEIBAUM: -- coming back around 550 Celsius.

MEMBER KIRCHNER: Yes. Yes. So then what about the concrete in the cavity. What temperate is the concrete steam?

MR. ZWEIBAUM: Let me ask our manager to answer this one if we --

MR. SONG: We did a --

PARTICIPANT: Who are you?

MR. SONG: Oh, sorry.

MEMBER KIRCHNER: It's kind of a leading question because you would have to stay below you, you know, your ACS or ACI. I forget the code.

(Simultaneously speaking.)

MEMBER KIRCHNER: American Concrete Institute limits. If that, if indeed, the chamber here is concrete.

MR. SONG: Yes. This is manager of steel structure, Brian Song. And yes, we are considering that. And considering to have the concrete temperature beyond the limit of ACI 39 that you described.

So that will be considered with the thermal management system, so we will, that is a consideration that we have.

DR. BLEY: This is Dennis Bley.

MEMBER KIRCHNER: Thank you.

DR. BLEY: On this sketch you show the feedwater valve closed. Now, you wouldn't have feedwater, but do you actually close it in case you do

have feedwater? This is covered in several different kinds of events.

MR. ZWEIBAUM: I mean, if we have feedwater available then we could be constantly replenishing the storage tank. We're not forcing that closed. But we are designing the system --

DR. BLEY: That's what I thought.

MR. ZWEIBAUM: -- to operate with it closed.

DR. BLEY: You're just saying there might not be feedwater and you're fine then. Okay.

MR. ZWEIBAUM: Correct.

DR. BLEY: Thanks.

MEMBER KIRCHNER: And just one follow-up question. This is Walt Kirchner again. The way you're showing the system there with the tank outside the primary, poor choice of words, whatever the reactor building is called, then this would be a safety grade, or a safety-related system and be hardened and protected against missiles, et cetera?

MR. ZWEIBAUM: Yes. The entire DHRS is safety-related.

MEMBER KIRCHNER: Right. So then that would have to be in a hardened enclosure.

MR. ZWEIBAUM: Yes. To be more precise,

the portions of the DHRS are required to perform the safety-related heat removal function, will be protected by the structure.

So, the feedwater portion, which is not required for the safety-related heat removal function may not necessarily be protected.

MEMBER KIRCHNER: Yes. No, I get that part. I was thinking of the tank itself.

MR. ZWEIBAUM: Right.

MEMBER KIRCHNER: For missile protection and seismic considerations.

MR. ZWEIBAUM: Yes. That is protected.

MEMBER BALLINGER: This is Ron Ballinger. I keep looking at this and I keep thinking t to the forth radiated heat transfer. And I keep wondering what kind of uncertainty might there be in all of this system because much of a change in temperate means a lot of changes in heat transfer.

I'm assuming that there will be an uncertainty analysis done of this whole system.

MR. ZWEIBAUM: Yes. Casey, do you want to take this one for more detail?

MR. TOMPKINS: Yes, sure. This is Casey here. So, because the temperature of the DHRS in our operation is pretty low, changes in that temperature

don't really effect heat removal too much. The temperature heat removal is mostly driven by the vessel temperature.

But in terms of the heat and defectors, there is uncertainty there, so we'll have to look at that. But for the most part we'll have correct test data on the anticipate heat removal from individual thimbles under prototypical cavity conditions that will give us higher confidence in what our removal rates are. And we have codings that we're looking into that give us more predictability.

MR. PEBBLES: And this is Drew Pebbles again. Just to be clear, these are forward looking statements right now for the PSAR. We don't have that level of detail in the application. But qualification of the system is done for the operating license application.

MEMBER BALLINGER: Thank you.

MEMBER KIRCHNER: This is Walt Kirchner again. Just a, this is probably a detail for the future. But with these thimble enclosures inside the cavity, the large flat plates that maximize the area and protect the concrete, were they just around annually, annulus structure of --

MR. ZWEIBAUM: They're around annulus

structures, but you have a fair amount that's around the vessel.

MEMBER KIRCHNER: Okay. Which four --

MR. ZWEIBAUM: We're obviously only showing one here but --

MEMBER KIRCHNER: Yes. Okay. Thank you.

MEMBER REMPE: You know, when I think about Ron's question and Dave's earlier question at the beginning of the meeting, this mockup that you're getting ready to build, can you use it?

I know it's not in the PSAR, but is there a vision that you might try and mock that up in that facility and quantify some of the uncertainties?

MR. ZWEIBAUM: Are you referring to the engineering test unit?

MEMBER REMPE: Yes. The one that's right before Hermes that looked like it was going to be the same scale, but when Dave was asking about surface heat transfer, or heat fluxes, I don't think we heard an answer to it. It's the --

MR. ZWEIBAUM: Right.

MEMBER REMPE: -- called the U-facility. That's what it's called.

MR. ZWEIBAUM: So the U-facility is after Hermes. The ETU --

MEMBER REMPE: Oh, you're right.

MR. ZWEIBAUM: -- is isothermal so we're not including the DHRS. But there will be a separate testing program for the DHRS to qualify it.

MEMBER REMPE: You're right. It's the one that was before Hermes I was asking for. But you have another test program that will be used for this? Okay, got it.

MR. ZWEIBAUM: Right.

MEMBER BALLINGER: Just --

CHAIRMAN PETTI: Can you --

MEMBER BALLINGER: -- there's got to be a lot of uncertainty. You know, plus or minus an inch, excuse me, 2.54 centimeters would make a heck of a difference.

CHAIRMAN PETTI: So just, I'm just wondering if you, you guys are probably aware of the tests that were done at Argonne for these types of heat removal systems. They did air. And then I believe they were going to do steam. Whether or not that geometry would be helpful here with what they're doing.

MR. ZWEIBAUM: Do we want to get into the details?

MR. PEBBLES: I think we'll take that

back.

CHAIRMAN PETTI: Okay.

MR. PEBBLES: But --

CHAIRMAN PETTI: Yes, I just, I don't know, I've lost track as to whether or not they got funded to do the steam. I know they did the air. But there may be something --

MR. PEBBLES: We did work with Argonne, we did work with that facility. And we are also planning on our internal campaigns to compliment that with more prototypic conditions.

CHAIRMAN PETTI: Ah. Okay, thanks.

MR. ZWEIBAUM: Okay, so --

MEMBER MARCH-LEUBA: Another question. All this is contingent on natural circulation of flow working inside the vessel. How much margin do we have on the Flibe volume?

I mean, how much inadvertent draining of the Flibe can you tolerate? I mean, I'm working the PRA here in my head and inadvertent drain of the vessel by a couple of inches will stop the, not the circulation and you're dead in tracks.

MR. PEBBLES: So the --

MEMBER MARCH-LEUBA: And I realize that you have a procedure to drain it, but the PRA should

have one branch on the tree to handle that.

MR. PEBBLES: So, a couple of points there. The detailed analysis wouldn't be until OOA. But for a PSAR we are committing to keeping the active core covered.

Nico mentioned the anti-siphon device, which does define a lowest level for the postulated event that we consider.

MEMBER MARCH-LEUBA: Right. But the preliminary potential conceptual cartoon design, I need to be convinced that you don't have inadvertent draining that stops the natural circulation. I need to be convinced that you have looked at it.

MR. PEBBLES: Yes. So if the core, if the fuel remains covered, than the path for natural circulation will be active. We are designing with that logic in mind since we have to maintain the fuel covered, in that condition the natural circulation test will be there.

MR. ZWEIBAUM: I'm also going to let Darrell Gardner weigh in here.

MR. GARDNER: So this is Darrell Gardner. I'm the senior director of licensing for Kairos Power. I think it's important, as I listen to conversation, lots of good questions.

Many of these questions are related to details of the design that are appropriate for a final safety analysis report. I think it's also important to remember that the findings at the PSAR stage are different than those at the FSAR stage.

And so, conclusions and determinations about safety acceptability are completely different. So while I understand the comment, I think we need to sort of pull back and remember what's required by the regulations at this phase.

MEMBER MARCH-LEUBA: So the regulations don't require the thing work?

MR. GARDNER: I'm sorry, I didn't understand the question.

MEMBER MARCH-LEUBA: The regulations don't require that the thing work safely?

MR. GARDNER: The regulations require that we describe the safety, that we describe the systems and the design criteria and the margins to safety. That's what's required to get a construction permit.

When we come back for the FSAR, the demonstration of how these things work is, that's where that demonstration is satisfied.

MEMBER MARCH-LEUBA: Okay.

MR. GARDNER: There is no, there is not a

determination of safety acceptance at this stage of the review.

MR. PEBBLES: Unless it's requested by the applicant.

MR. GARDNER: Unless we request that. And we have not requested that the staff make a determination of safety acceptability at this time.

MEMBER MARCH-LEUBA: Okay. And I will make sure that the ACRS letter says that in the first paragraph. That we have no idea about the safety of these reactors.

MR. GARDNER: Well, I'm not sure I would necessarily agree with that comment. I think I would suggest that you may not know all the details of how it's satisfied at this stage.

MEMBER MARCH-LEUBA: You're -- I'm giving you a, have you thought about this possible accident in your conceptual design and you're telling me to get lost. So I receive your comment.

MR. GARDNER: I don't think we're saying that at all. I think we're trying to set the framework for the questions that need to be resolved at this stage of the review versus at a different stage of the review.

MEMBER MARCH-LEUBA: I'll reserve my

questions --

CHAIRMAN PETTI: Fair enough.

MEMBER MARCH-LEUBA: -- for the Staff.

CHAIRMAN PETTI: Fair enough at this point. Can I just ask a question? In terms of your slides, A, we need a break. We'll also need the Staff to talk. Where are we in terms of slides left?

MR. PEBBLES: So, two/thirds into it.

CHAIRMAN PETTI: Two/thirds?

MR. PEBBLES: Yes. We had, we have another hour's worth of slide material.

CHAIRMAN PETTI: Another hour. That would put us at 3:00. Okay, let's just keep going. We'll also probably want a break. I thought the natural break would be between you guys and the Staff, but so, Members, if anyone feels like we need a break before that, let me know. But let's just keep going because I fear we are falling further behind. Is that probably true?

MR. PEBBLES: Yes.

MR. ZWEIBAUM: Yes.

CHAIRMAN PETTI: This always happens, so.

(Laughter.)

CHAIRMAN PETTI: We're just so interested in all the details. Let's keep going. Thanks.

MR. ZWEIBAUM: Okay. Next system is changing gears a little bit, but this is about the pebble handling and storage system.

Again, our fuel comes in pebble form. So this is really the system that handles, moves the fuel around and stores it. From initial onsite received through in process circulation down to final onsite storage.

A number of key sub-systems, one is the pebble extraction machine that sits on top of the reactor vessel and extracts pebbles from the core. This is a single screw mechanism that removes the pebbles from the molten salt.

We have a pebble inspection system that will perform flaw detection and burn-up measurement of removed pebbles. We have a processing system that will sort pebbles into appropriate buffer storage channels based on pebble types.

We have an insertion system, which is a separate wheel feeder mechanism that inserts pebbles back into the reactor via an in-vessel insertion line.

We have a number of storage system canisters. Each canister can store around 2,000 damaged or spent fuel pebbles in a non-critical configuration.

We have a storage cooling area. That area will be passively cooled in building for spent fuel canisters. And we have a new pebble addition system which stores fresh fuel and prepares them for circulation via high temperature bake out.

The next slide is a set of animations to show you kind of the journey of the pebble through the system. So if we start at the bottom right is where you can see this rough diagram of the reactor core. That red dot here would be a fuel pebble.

If we go next, that pebble goes up the pebble extraction machine through an off head penetration down to the inspection station that I mentioned earlier. At that stage the pebbles are inspected.

We make the determination between fuel and moderator pebbles. If a pebble is a moderator pebble then next it gets sent to a moderator storage bin.

If it's a fuel pebble next it goes to the burn-up measurement station. This allows us to know what the burn-up level of the fuel is to know if we're below or above the threshold where the pebble has reached its end of life, or effective life for Hermes.

Then next that pebble would get into processing. And next into buffer storage.

Now if we, next, we have other pebbles, and that could be of any different type, then that could also get stored into a number of other buffer storage canisters.

And the last case, next, would be if, through burn-up measurement we find that this is a spent fuel pebble then it gets discarded and sent into one active storage canister that is connected to the PHSS inner gas boundary at all times.

Next. If you look at the bottom towards the bottom right you can see the new pebble insertion canister. This is where new or fresh pebbles would be stored. Whenever we send pebbles to active storage we insert new pebbles.

Those go through the same inspection station that recirculated fuel goes through. In case we can detect any flaws then those pebbles would be discarded immediately. Otherwise, they get processed. And next go into one last storage bin that would contain pebbles with no burn-up. Essentially fresh fuel.

CHAIRMAN PETTI: So can I ask a question? What differentiates a new pebble from a moderator pebble in terms of the inspection?

I'm assuming you were using gamma to

determine moderator from fuel, from irradiated fuel.

MR. ZWEIBAUM: Gareth, do you want to take this? We have our, the responsible engineer for our pebble handling system.

MR. WHATCOTT: Sure. No problem, Nico. When we insert new fuel we will do those in sort of a sequential fashion. So we'll be able to know that this line of pebbles coming in are all new.

CHAIRMAN PETTI: Ah, okay.

MR. WHATCOTT: That way we can maintain an inventory of how many pebbles we've introduced.

CHAIRMAN PETTI: Okay.

MR. WHATCOTT: To differentiate between a moderator and fuel pebbles, you mentioned gamma. That's certainly one option. Another option we're looking into currently is temperature since moderator pebbles won't have decay and so they should be thermally at a different temperature.

CHAIRMAN PETTI: Ah.

MR. WHATCOTT: And we can detect that earlier on before having put it through a gamma spectrometer.

CHAIRMAN PETTI: Yes. I mean, you probably are aware of, in pebble beds, this measurement is critical and is not as easy as it

sounds or looks in a simple diagram like this.

You have, at least for the pebble beds that I was aware of, you had like 30 seconds to make the measurement. And signal to noise ratio, looking for the cesium-137 peak, which is generally a good strong peak, but in fuel with all the other stuff it's not as easy as it sounds.

So anything, another measurement could be quite, quite useful in case that one is difficult. Yes.

MR. WHATCOTT: Yes. No, we certainly recognize the challenge with making this burn-up measurement. We engaged with Sandia National Laboratory and are working on some experimental work with them to make sure that we can, we provide enough time to develop this technology because as you mentioned, other pebble bed systems have shown that it's a challenging measurement to make on freshly removed fuel that has high radioactivity.

So, yes. Looking at, using something like a thermal, a thermal camera to screen out pebbles is something we'd like to do as well so we're not having to scan moderator pebbles and waste that time of the gamma specter spectrometer.

CHAIRMAN PETTI: Yes.

MEMBER REMPE: Dave, I had a couple of questions here. And I know we're behind so answer what you can in a hurry and save the rest of them for later.

But first of all, the construction permit indicates that the canisters for the spent fuel storage pebbles are flooded. And then it also talks about during a full core offload that the pebbles aren't sorted, you just put them in a canister.

And I was curious whether those canisters would be flooded, and if so, then I am curious about how you dry them out. And if you're going to measure the off gas, then how that system is going to work in any details of interest.

And then, finally, I believe NEIMA requires that folks think about the whole fuel cycle. And I was curious about what you will ultimately do with these canisters of pebbles.

Back in the GA days they talked about pushing the rods out of the fuel assemblies to try and reduce the volume of the waste. I'm not sure what you do with the pebbles. Maybe you even know, Dave, what they've done in Germany. I've seen articles where people talked about trying to break them and separate the particles from the graphite to reduce the volume,

or is this not part of the Kairos plan to think about what they're going to do with spent pebbles?

MR. ZWEIBAUM: I'll just answer the first question because I think is very relevant to the current conversation and safety piece. I think as far as the fuel cycle we might table that to a later discussion.

So, to clarify, the idea of flooding, if I know what you're referring to in the PSAR, is purely based on the analysis that was done to ensure that we don't have any critical configuration. So in a worst case scenario, where the canisters would be flooded, we are conserving that we are still not in a critical configuration.

That being said, this is not an intentional flooding of the canisters. The canisters, I think the current baseline is that they would be stored temporarily in a storage pool, but the canisters would be sealed. So we wouldn't have any water ingress into those canisters.

The flooding, in the context of the PSAR, was only related to the analysis that was done to ensure that we don't have any critical configuration of the fuel at any point.

MEMBER REMPE: Okay, so that helps. So

this would be a rare event, and if you did have to do something with those pebbles it would be a rare event and it's not something that's a planned operational thing where you have to worry about drying out pebbles that --

MR. ZWEIBAUM: That is right.

MEMBER REMPE: That helps.

CHAIRMAN PETTI: Okay --

MEMBER REMPE: And then at some point I am curious about what you're going to do with the fuel from this when it's --

CHAIRMAN PETTI: So I can just tell you, Joy, the Germans just, in AVR and THTR, they didn't do anything to the pebbles. And it's, you know, it's the one thing that a prismatic has as a benefit is you can take the compacts out and reduce the volume.

They didn't do anything to burn the pebble matrix off or anything. So they just had a really large volume of waste to deal with.

MEMBER REMPE: But it was only a couple of reactors and I assume people want to do more than one or two reactors with this. And so, again, we need to think of the whole fuel cycle now because I guess congressmen folks put that in the bill when they were thinking about these new reactors.

MR. ZWEIBAUM: Yes. The end of the pebble journey, for those of you who wants to know, the pebble insertion hopper that is sitting at the top of the reactor vessel, that would take the recirculated fuel, take it down a pebble insertion line and the pebbles are reinserted through the bottom of the core. And repeat. So that is it for the PHSS.

CHAIRMAN PETTI: And you do plan to mock this up, right, somewhere along the line?

MR. ZWEIBAUM: Mock this up as far as physical testing?

CHAIRMAN PETTI: Yes. Yes.

MR. ZWEIBAUM: Yes. We've done a number of scaled tests already. And we will continue to do that. This is in scope for the engineering test unit. And we'll have a number of other tests to confirm those processes.

CHAIRMAN PETTI: Great. Thanks.

MR. PEBBLES: All right. So just real quick, Dave, we just wanted to check with you and see if this is where you wanted to take the break or if you wanted to wait till the end?

CHAIRMAN PETTI: No, let's at least get through yours.

MR. PEBBLES: Okay. All right.

MR. SONG: All right, so I'm Brian Song, I'm manager of civil structures. And I'll go over the Hermes civil structural stuff. Mainly Chapter 3 of PSAR.

So, as you can see here, the reactor building is approximately 250 feet long and 100 feet wide. And the philosophy here is to design, to separate the design and decouple the safety-related portion of the building and the non-safety-related portion of the building, which contains the SSCs, to consolidate protection.

The safety-related portion of this building is approximately 180 feet long and 50 feet wide. And the design strategy of modulated inflexibility is considered to allow for speed of construction. And it is a, we are trying to make that as simple as possible. So, it's a simple configuration.

The safety-related building structure uses a based isolation. And the non-safety-related building is surrounded of the isolated super structure.

The safety-related reactor building base slab is approximately at grade with isolator basement below. And the foundations are transferred to loads

of stiff rock.

The safety-related structure and reinforced concrete structure that is hybrid of casts in place, and also pre-cast concrete structure elements. The safety-related person is designed to protect safety-related SSCs from internal and external events. Including potential damage from the non-safety-related portion of the building.

To credit, safety function, safety-related reactor building is to protect and support the safety-related SSCs. And is not confinement or containment.

The building is applying performance based design principles to align criteria with credited safety function.

As you can see here, the safety-related portion of the reactor building is divided into cells. And the cells contain all the safety-related SSCs in the facilities. And also some of the non-safety-related SSCs.

And I think there was a question about the DHRS, so the DHRS is included in the reactor building cell that you see here.

The message related portion of the building is comprised of maintenance halls, including high-bay shelves, maintenance corridors, truck bay

auxiliary worker inhabited areas. And it is a steel frame construction with an independent foundation system that will consist of a max slab with grade beams.

And the non-safety-related portion of the reactor building does not contain any safety-related SSCs. And this portion of the building is designed so that the payload does not interfere with the safety functions of safe SSCs located in the safety-related portion of the building. Or, yes. That's what, yes. So that's kind of what this slide is. Any questions?

MEMBER BALLINGER: Yes. So, this is Ron Ballinger. So the moat is just a separator?

MR. SONG: The moat, the moat is, yes. It has a -- so the moat wall has two functions. So because it's base-isolated so it will protect from the displacement of the safety-related portion. And yes, it is a separation of the building.

So a separation from the safety-related to the non-safety-related portion of the building. Yes.

MEMBER BALLINGER: So is it seismically significant?

MR. SONG: So, currently we're considering this not to be safety-related. Maybe I will ask one of my subject expert, Ben, if you want to add on to

the comment there. Ben Koslow.

MR. KOSLOW: Sure. Thanks, Brian.

MR. SONG: Yes.

MR. KOSLOW: The moat is sized so that under the ground motions inspected for the site that we have ample physical space so that the safety-related building does not come in contact with the non-safety-related building. And then any of the distribution systems that cross that gap have adequate flexibility to accommodate that expected deformation as well.

MEMBER BALLINGER: Does it have any function to deal with thermal expansion because you're dealing with very significant temperature differences in various parts of that building?

MR. KOSLOW: So by the time you get out to the moat, which is a fair distance away from the reactor, it's not anticipated to have extreme temperature fluctuations.

MEMBER BALLINGER: Okay.

MR. KOSLOW: Certainly the temperature profile is accounted for when sizing things.

MEMBER BALLINGER: Thanks.

CHAIRMAN PETTI: Sometimes people might call the moat a seismic gap. I've seen that in other

designs.

MEMBER BALLINGER: That's what I was kind of thinking is what it was.

CHAIRMAN PETTI: Yes. Yes.

MEMBER KIRCHNER: Yes.

CHAIRMAN PETTI: So let me just ask a --

(Simultaneously speaking.)

MEMBER KIRCHNER: Dave, can I ask a question?

CHAIRMAN PETTI: Yes, go ahead.

MEMBER KIRCHNER: This is Walt Kirchner. Are you planning, for the safety part of the building there, the inner part that's isolated, are you planning on using steel plate composite concrete construction?

MR. KOSLOW: Currently we're not. We are considering precast concrete and cast in place as a hybrid. However, that can be considered during our design iteration we might, we'll see if that is appropriate.

MEMBER KIRCHNER: It's just that this is just an observation, not a request. It's just one member.

You might look at that as an option for the building that's isolated in terms of just ease of

construction vis-a-vis casting concrete and isolating the concrete cast, complete cast structure. It's just a thought. It's an observation, it's not a request.

MR. KOSLOW: Yes. Thank you for that.

MEMBER REMPE: Dave, I have a question too on this building layout before you switch. Oh, Dave, excuse me, I have a question about the reactor building layout on the prior slide.

When I look at the CP application on saw in Section 9.8.1 that they mentioned that there is going to be a hot cells and a PIE and materials testing laboratory facilities in the, could you tell me where those are located? Are they in this building?

MR. SONG: So currently the layout, it's not -- so this is more preliminary based on the image that is in the PSAR. That I don't think we actually located that yet.

MEMBER REMPE: But it will be in this building somewhere, is that a true statement? Or is it going to be in a different building? Or are they going to be in a different building?

MR. SONG: So, what was the system again? Sorry, I --

MEMBER REMPE: Well, in Section 9.8.1 of

the PSAR it says that there is a hot cell, there is a PIE and materials testing laboratory facilities. And I was curious where they were all located.

They did say, I believe, that there was a crane located, associated with them, and I was just kind of curious where all of these facilities are. I guess we can save the question till later, but I'm just curious because, again, is there a potential you could have any sort of radiation releases in the hot cell facility.

How do you get stuff from the reactor to the hot cell? I mean, there's a lot of those kind of questions that we'll have to be thinking about as we go through this review.

MR. KOSLOW: Yes, we'll double check the words in the application and get back to you later in the meeting.

MEMBER REMPE: Thank you.

MR. SONG: Thank you for the comment. So the design considered is meteorological loads, such as rain, snow, wind, tornado, and windblown missiles for the site per local building code and NRC guidance for the site.

The safety-related building is designed without crediting the non-safety-related exterior

shell for protection from snow, wind, rain, and missile loads. The exterior shell of the safety-related building is designed with a thickness to protect the safety-related SSCs from high-wind missiles, including debris from potential damage of the non-safety-related reactor building.

Flooding loads. The safety-related dumping will be protected from internal and external, I'm sorry, internal flood with shields, curves and drains, et cetera. Safety-related reactor venting protects the safety-related SSCs from credible external flood.

And the external envelopes uses water tight flood protection features as well. There is also isolator in the basement. However, the maximum credible flood elevation is higher than that, but the isolators will still perform with their function. What's being provided.

So let's go to the next slide please. For seismic loads we have been using risk-informed performance based insights to determine the seismic design criteria. For instance, ASCE 43-19 and we define SDC 3 to be the criteria for the safety-related SSCs.

The seismic design basis earthquake is

based on site specific seismic hazards considering other recent nearby seismic hazard analysis. And site specific geotechnical characteristics. And we will confirm the data in the OOA.

The safety-related reactor building incorporates spring dash pot (phonetic) seismic isolation system, which lowers seismic demands of the safety-related building and safety-related SSCs in both horizontal and vertical directions.

And the moat wall and the flex connections are considered to accommodate the displacements of the isolated safety-related building. And also the -- and the safety-related portion of the reactor building will be represented by a three-dimensional FEA developed in accordance with Chapter 3 of ASCE (audio interference).

So that's -- I think that's --

MEMBER KIRCHNER: Another question, may I, Dave? This is Walt.

Is the safety-related portion of the reactor building also a functional containment or confinement or does it --

MR. SONG: No, it's not. Yeah.

MEMBER KIRCHNER: So, where do you protect it against leaks of Flibe in the system? At that

inner cavity is the place where you isolated?

For example, you showed isolation valves on penetrations for the Flibe fluid lines. You show a penetration for the pebble introduction and removal systems. I didn't see isolation valves on that.

Where do you -- what do you try and control any kind of fission product or Flibe leakage, where is that done in terms of creating a "like a confinement boundary" where you can control the atmosphere because you're dealing with a toxic material?

MR. HAGAMAN: This is Jordan Hagaman, director of reliability engineering. In terms of Flibe, the primary priority is to preserve enough Flibe in the reactor vessel itself to maintain the cooling function.

With regard to non-nuclear safety considerations, we haven't discussed any of that in a preliminary safety analysis report. And we're not using the building for any physical confinement functions for nuclear safety.

MR. GARDNER: So this is Darrell Gardner. I would add one more thing there, that I think the question you're asking really is more along the lines of what we would consider contamination control, it's

not a confinement function.

MEMBER KIRCHNER: Okay. But it's a, let me back off functional containment and just say confinement. But some control, wouldn't your design philosophy try and have a, how should I say it, a barrier such that if you had fission product leakage and/or leakage of Flibe, that you would have some ventilation capability to have ventilation that would be effective. It drives you to have kind of a minimal leakage from that reactor building, right?

MR. GARDNER: Darrell Gardner again. Just to reiterate, it would be from the contamination control perspective, not from a dose consequence to the public perspective.

MEMBER KIRCHNER: Okay.

MR. GARDNER: The Flibe is retaining the fission product. It's --

MEMBER KIRCHNER: Yes. That's under normal operation considerations. I'm just, I'm trying to think of the fact --

CHAIRMAN PETTI: But the tritium --

MEMBER KIRCHNER: -- you have leakage the tritium is going to come out.

CHAIRMAN PETTI: You've got ventilation, right? And you've got a, what do you call it, traps

to, if there is any tritium in the vapor, in the building.

MR. GARDNER: That's correct. There is a non-safety HVAC system. That's why I said, this is part of contamination control and effluent control, it's not a dose consequence control.

MR. ZWEIBAUM: So before we move on, I did want to circle back to Dr. Rempe's question about Section 9.8.1. I did look at that section. And we do say that that system is located in the reactor building.

So like Brian said, the specific location in the reactor building hasn't been nailed down, but it is in the reactor building.

MEMBER REMPE: So thank you for that clarification. I guess then I'm kind of thinking that as part of a construction permit evaluation that it would be behooves to know where it's located and how material would come from the reactor in to some of these facilities because that, I would think, would fall under the purview of what we're thinking about since you're pouring concrete to accommodate this and we're supposed to understand the safety margin, I believe, associated with some of these things.

And although we won't have all the

details, it seems like we need to know where it's located. And there is a pathway that's adequately protected. Does that seem reasonable to you?

MR. GARDNER: So, this is Darrell Gardner. I'll speak to that quickly. Again, I think the kinds of things that you're broaching over into is worker protection and Part 20 requirements. Which are not traditionally part of the PSAR.

We have addressed some bounding effluent considerations, but as we note in the application, those sort of details on things like shielding and contamination control will all be addressed in the FSAR.

MEMBER REMPE: Okay. Again, the Staff will help decide this I guess too, but I'm thinking of what we've seen in other construction permits we've reviewed for other NPUFs in recent times. And we at least kind of knew where the various rooms were located in the building for some of the processes involved. But we'll explore that further as we go along in this review.

MR. GARDNER: Okay.

MR. CILLIERS: I'm going to go. Hi. Thank you for the opportunity. This is Anthonie Cilliers. I'm director for instrumentation controls

and electrical.

I'm going to start leading us into the principles that we used to design the system and then we'll move on to have a look at the architecture. I'll also try and address some of the questions that has come up before throughout the presentation, and then we can have a further discussion.

Our I&C system is very much designed based on the primary functions of the KP-FHR technology of the reactor that we are designing. These include features like a system so there is no depressurization when you trip. And there's a large heat capacity in the coolant. And pretty slow transients changes inside the reactor itself. And of course, large safety margin for the fuel integrity as well as for the coolant. And these features are very important for us when we were designing our reactor protection system.

We have separated our I&C system into various areas. First, we have our reactor protection system. And they have been very deliberate to detect an act on the fundamental metrics that might challenge the integrity of the key system structures and components.

And it relies on shutting down systems.

So it does rely on active power systems to shut it down. So it relies basically on detect and shutdown of various systems.

Then the next one we have is the plant control systems, which is a non-safety-related system. In this area this system relies of exhaust of hosts of additional instruments throughout the plant. And that is used to control plant operation, as well as early detection of component failures to act on that before we move into the safety space or any of the safety, the SSCs integrity is challenged.

And we also have an intelligent health monitoring system building. Again, non-safety-related, another safety-related system. This system uses computational technics combined with instrumentation data to detect component degradation over time to assist us with operations and maintenance. And also, keep us further away from safety scenarios.

There has been questions about the instruments specifically. Our instruments for the plant, for the reactor protection system include discrete level sensing in the core itself. That is an in-house development process that we're going through for custom detection working with the Flibe itself.

We will use thermal measurements in the core. And again, we've been very deliberate in the PSAR not to specifically talk about specific technologies as we are evaluating various options in these areas.

Neutron flux measurements as well is inputs to the reactor protection system. At the moment we are analyzing the exact location of them, so you would have seen that some locations already specify these locations to be specified.

But we have determined that our power range detectors will most likely be outside of the core. Almost definitely be outside of the core. So is range detection is still, is still something that we are finalizing.

And then we will also have indication for a break in the pebble handling line, which is not specified in the PSAR currently, but specific measurement we'll be using there. But we have a couple of different options that we are evaluating.

I think we can go to the next slide from here.

MEMBER MARCH-LEUBA: Sorry.

MR. CILLIERS: Yes.

MEMBER MARCH-LEUBA: This is Jose again.

The use of ex-core detectors will be outside the core but inside the vessel in the --

MR. CILLIERS: Outside the vessel. Outside the vessel, yes.

MEMBER MARCH-LEUBA: Outside the vessel where the DHRSs are?

MR. CILLIERS: Yes. Correct. It's with the bioshield. Yes, likely in the bioshield area outside of the vessel.

MEMBER MARCH-LEUBA: We still don't have all the nomenclature of everything you call, but, so it's not going to be in the reflectors it's going to be outside --

MR. CILLIERS: It's not going to be in the reflector, it's going to be outside the structure of the vessel itself.

MEMBER MARCH-LEUBA: Very, very far away from the core?

MR. CILLIERS: Yes. Yes. And the temperature is also indicative of that, the temperatures that those instruments will see. So it's much lower temperatures that will be exposed to, compared to what they will, that instruments inside the core will see.

MEMBER MARCH-LEUBA: And this core has a

relatively low power auxiliary. Have you done some estimates that you have sufficient signal to have to drive the detectors?

MR. CILLIERS: Yes, that's correct. That's where -- that's how we decide on the exact location of these detectors.

Of course the source range detectors is a little bit different because it also relies on the size of the source itself and the location. But we are very encouraging information there that they could also probably be moved outside of the vessel itself.

CHAIRMAN PETTI: I'm glad you said that --

MR. CILLIERS: But the analysis --

CHAIRMAN PETTI: -- because I was going to recommend that you look at that specifically. I think in these small cores you could even put source range stuff outside of the --

MR. CILLIERS: Yes.

CHAIRMAN PETTI: -- reactor which would simplify a lot of things.

MR. CILLIERS: That's correct. So yes, we will plan that out once we have more information on the analysis from our mod safety (phonetic).

MEMBER MARCH-LEUBA: And this is Jose again.

MR. CILLIERS: Yes.

MEMBER MARCH-LEUBA: On the cartoons earlier we saw only three detectors for both level and power.

MR. CILLIERS: Yes.

MEMBER MARCH-LEUBA: Have you decided that you're only going to have three protection channels?

MR. CILLIERS: No. At the moment we are deciding between two and three. You will see in the Chapter 7 of PSAR we always expect four channels. So most likely we will move them outside to have four of them.

We are planning to have neutron flux mapping inside the core. And those will not be safety-related instruments that may use only the three detectors.

MEMBER MARCH-LEUBA: Those are equivalent to the SPDMs or the LPRMs in BWRs?

(Simultaneously speaking.)

MR. CILLIERS: The flux mapping.

MEMBER MARCH-LEUBA: Flux mapping. In BWRs they're called LOCA power range monitors.

MS. CROWDER: Yes.

MEMBER MARCH-LEUBA: In PWRs they're called SPDMs.

MR. CILLIERS: Yes. So, yes, that's for flux mapping to determine what the flux shape is inside the core.

MEMBER MARCH-LEUBA: Okay. Okay.

MEMBER BROWN: Can you back a slide?

MR. CILLIERS: Can I continue?

MEMBER BROWN: Yes. What do you mean a --

MR. CILLIERS: Okay, next slide.

MEMBER BROWN: No, no, no, no.

MR. CILLIERS: Oh, sorry.

MEMBER BROWN: Go back to --

MR. CILLIERS: Okay, I'm sorry.

MEMBER BROWN: What do you mean by semi-autonomous control room?

MR. CILLIERS: Oh, I apologize, I should have continued there. So our control room does not have any, we do not create any safety functions by operators. So the control room itself acts as a view into the reactor and into the plant itself.

So the operators are operating the plant under normal operational conditions. But it does -- and some of the functions of operations is automated.

But the reactor protection system acts completely separate from the control room itself. So we are moving away from the relying on operator

actions for safety functions.

MEMBER MARCH-LEUBA: So what you're saying is that the operators are not relied upon for any safety function?

MR. CILLIERS: None at all.

MEMBER MARCH-LEUBA: And can you --

MR. CILLIERS: In the same way we would --

(Simultaneously speaking.)

MR. CILLIERS: Yes, go on.

MEMBER MARCH-LEUBA: Yes. Yes, then you would have to worry about the force inside the core. Anything the operator can do to make it go bad.

I mean, there are errors of omission and errors of commission. Have you considered those?

MR. CILLIERS: Sorry, you have to repeat that, I --

MEMBER MARCH-LEUBA: There are some things called errors of omission when the operator doesn't do something. And then this error of commission where the operator does the wrong thing.

MR. CILLIERS: Yes. I will explain in the next slide what the principles are of that.

MEMBER MARCH-LEUBA: Okay.

MR. CILLIERS: And then of course on the electrical --

MEMBER REMPE: Excuse me. Are you planning to have the operators licensed by the NRC?

MR. CILLIERS: At this stage we are considering that, but we have to consider what is actually the functions of the operators to have them licensed. But yes, we are working through how they will be licensed.

MEMBER REMPE: Thank you.

MR. CILLIERS: Then the last point there on the electrical supply system, we'll have a slide on that architecture. It's important to say that we, to state that we do not have safety-related electrical supply because we do not rely on the electrical supply for any safety functions. And so that's outside of the safety-related scope as well.

MEMBER MARCH-LEUBA: Are you planning to have battery backup for monitoring?

MR. CILLIERS: We do have batter backup. And I'll show that when we get to the layout of the --

MEMBER MARCH-LEUBA: Okay. I'll wait then.

MR. CILLIERS: -- electrical system as well. Although it's on safety-related.

So here is a little bit of a, the principles that we have. As I mentioned, we are very

deliberate in the fundamental matrix of what we want to measure for the reactor protection system so that we can predict the integrity of the system itself.

You'll see on the left-hand side that's a depiction of a operating envelope. The blue area is the pump control system. And that is where the operators is acting within.

And that brings us to the point where the question came from the omission of operators or the operators deliberately or by accident doing something incorrectly. As long as the operations, both operational parameters remain inside that blue, the light blue area, the reactor protection system does not intervene and the operators can actually operate and make mistakes if that should happen without challenging the integrity of the system itself.

Once it crosses the boundary into the red space, the reactor protection system, those very specific measurements, metrics will determine that the plant is now in a space where it could challenge the plant integrity. And automatically the reactor protection system will intervene and shut the reactor down.

At the same time it will also block all operations coming from the plant control systems. So

it will take over complete control of the system. And I will show you in the architecture how that is done.

Now, you'll see on the right-hand side there is a graph of the different temperatures that we're looking at and different instruments and that will operate in specific ranges.

An important point to take away there is our fuel integrity is 1,600 degrees. It's a very, very high temperature, and we're staying far away from that. Flood boiling point temperature is 1,430 degrees. Again, very far away from that. As well as the vessel integrity, which is 850 degrees.

So our whole analysis of early detection of operations that could challenge that revolves around staying far away from that temperature as well. And so we'll be operating in the range below 700, 705 and above 460 degrees.

And the real principle of the operating system, of the plant control system is to maintain our operating parameters in that blue boundary. And if it crosses over at any time that's where we will check the reactor. Any questions on that?

MEMBER BROWN: Yes.

MEMBER REMPE: Yes. Go ahead, Charlie.

MEMBER BROWN: Basic reactivity control

and plant condition control, is that done by the operator?

MR. CILLIERS: Yes. That can be done by the operator inside the blue space.

MEMBER BROWN: I heard you use, you said can be.

MR. CILLIERS: Well they can be. The plant control system will be automated to maintain that at certain levels. But the operator will be able to control the reactivity inside the blue space. Yes.

MEMBER BROWN: So, startups are done automatically. You punch a button, the plant starts up and everybody goes to sleep, is that the way they envisioned it?

MR. CILLIERS: No, that's not the way we envisioned it.

(Laughter.)

MEMBER BROWN: I'm being --

MR. CILLIERS: We will have a step-by-step approach to get the plant up to temperature. And as soon as we are in the blue space, then the operators will be able to operate it. When the plant is heated up and we can go to criticality.

MEMBER BROWN: Is the plant operating condition maintained automatically without any

operator input? That's all I'm saying. Once you get into the blue space --

MR. CILLIERS: Yes.

MEMBER BROWN: -- is it hands off?

MR. CILLIERS: Not in the blue space. In the blue space the plant conditions will be automatically controlled. But the operators can change those conditions. As long as they remain within the blue space.

CHAIRMAN PETTI: But they don't have to, right?

MR. CILLIERS: They don't have to, no.

CHAIRMAN PETTI: Okay.

MR. CILLIERS: We could change modes. There is a number of modes that it can move into. But the operators, as long as they're in the blue space the operators do have autonomy to be able to make decisions in that space.

And it's important to note, once they get into the red space the operators do not have any control over what, to change any of those parameters. And I'll explain that. I have a couple of examples in the next slide.

MEMBER REMPE: So maybe you're planning to mention this in the next few slides, but in Table 7.2-

1 it lists some control parameters. And I just was curious why the coolant level isn't listed as a control parameter?

MR. CILLIERS: The coolant level within the vessel itself?

MEMBER REMPE: Right. Because it seems to me, again, you want to keep that coolant above the core for natural circulation, as well as for fission product retention.

MR. CILLIERS: Right. I will have to have a look at that. I think, I believe the coolant level is maintained through the syphon system, as well as our full drain system. And then of course we've got the trips on low and high level of the coolant. But I will have to have a look, because I think -

MEMBER REMPE: Okay. Maybe I missed some things too because it seems like it ought to be there. Thank you.

MR. CILLIERS: Yes. Okay. So the next slide you see, this is our architecture. And you see in the PSAR as well.

You can see the separation between the blue areas, which is non-safety, and the red areas, which is the safety system. It's really important to note that they are absolutely isolated from one

another. And this is achieved in two ways.

The reactor protection system in red communications information out through a one-way, we call it a data diode. But the reality is, there is only one-way communication out of the reactor protection system. It does not have hardware that can take signals from the outside.

So it's a one-way communication to the operators so that they can see what the indications inside the reactor protection system. The reactor protection system knows about itself and the indications that it's getting from the reactor, which I mentioned before includes the discrete level indication, temperature indication, neutron detection, as well as the DHRS line break. Those indications are used for trips.

There is another output from the reactor protection system, which is a slightly different one from normal trips. And that is the DHRS activation. And I think Nico explained some of that earlier on.

And what happens with DHRS is, once we've reached a certain power level or accumulated a certain level of fission products in the core, based on the count of the neutron detectors the DHRS will be activated by the reactor protection system

automatically and it will be locked so that the operators are unable to deactivate it.

After shutdown, should a shutdown occur, if the temperature is, goes low enough, that the reactor protection system will then release that lock so that the DHRS can be deactivated. But although it won't be deactivated automatically. It will just allow deactivation. So that's a spatial nuance to what the reactor protection system does.

As for the rest, the reactor protection system, it measures level temperature flux. And based on that it activates the RCCS control safety elements that will drop into the reflector, into the core based on their design using gravity. So it actually removes power from the system to drop those rods.

It also removes power from all the active systems that the plant control system is operating. That includes the primary salt pump, the PHSS, the pebble handling system, as well as the other flood coolant systems.

It's very much designed around using the DHRS as our active cooling system. Having said that, this is when we reach that great levels in the operating envelop, the plant control system in the blue space has got access to a whole host of

instruments.

And should anything, any failure be detected through those instruments, the plant is also, can also be shut down, or power can be reduced as necessary, long before we reach the level where the reactor protection system is required to intervene. Any questions on that?

MEMBER BROWN: Absolutely. I notice in your diagram that all your main plant control systems, all your non-safety control systems, are directly connected to the internet, through gateways and ethernet connections. So a hacker can come in and turn your plant upside down.

MR. CILLIERS: That is a really good question. We talk about that quite a lot. It is -- we are all feeding information -- or the plant will be feeding information to our support systems outside. We rely heavily on the required cybersecurity features that we're building in, but the same scenario, if you go one slide back maybe, the -- as long as you stay in the blue side, the blue space, the reactor cannot be turned upside down. The reactor protection system will always intervene if something like that should happen.

Go on to the next slide. And you can see

there that the rate system is not connected to any part of the outside world whatsoever. That being said, we are not taking cybersecurity lightly, and implementing all the needed cybersecurity implementations of making sure that -- yes?

MEMBER BROWN: You could make it really clear if you take that cloud away and don't put anything in there. You can assess that issue when you get down to the details.

MR. CILLIERS: Yes.

MEMBER BROWN: Very heavily detailed. That is -- I probably shouldn't say anything, but to me, that's totally unacceptable. I'm just passing on one member's conclusion from reading this.

MR. CILLIERS: So noted. Thank you.

Next slide. Okay, I think this is a bit of animation, so you can go to the first animation.

MEMBER BROWN: Can I give you one other observation before you go on?

MR. CILLIERS: Sure.

MEMBER BROWN: You can go to -- is this the picture, the next slide?

MR. CILLIERS: That's the next slide, it should show pretty much the same system.

MEMBER BROWN: Yeah, well, you have -- the

one I've got, Slide 39, shows pictures of operators sitting at screens.

MR. CILLIERS: Yes, that's coming up. So that's the animation.

So, okay, just to give you an example of the path that we are going through, with our engineering test unit, we are actually building all of these systems and testing them with all the features that we require.

So the first animation, that is our flood control system. That is actually being tested right now. That includes controls of the primary salt pump and various others, so basically all the non-safety features.

Next slide.

MEMBER BROWN: So they're all integrated?

MR. CILLIERS: At the moment, they're all integrated, yes.

MEMBER BROWN: So one or two processes integrate all the plant-controlled functions into those areas, totally? Even though they're non-safety-related, you've totally integrated that system? So if a box goes up in flames, you're toast?

MR. CILLIERS: Yes, but for the reactor system as well, that we will not integrate it in that

way. So the functions are separate, but they run on a single box.

MEMBER BROWN: I saw that in your non-animation. Go ahead.

MR. CILLIERS: Okay. So, at the bottom you can see our instrumentation test unit. That's where we are testing all our salt-wetted instruments at temperature and that's being conducted right now.

Next slide. The next one. That is our remote support room where we are sourcing data through -- from the control system where -- that's located in our headquarters here in Albuquerque and that's supporting for the engineers to support the operations of the system.

Next slide. That's our project control room that's in Albuquerque, New Mexico, where we are developing the human-machine interfaces for the system itself, as well as connecting that to the simulators to test all our operations.

Next one. This is a couple -- just a picture of the different instruments that we are actually implementing in our engineering test unit. You can see the little green one at the bottom right corner. That is our in-house-developed level switch, which uses two probes that, when the Flibe touches the

two probes, it indicates a level indication. So it operates really fast and it's custom made for the Flibe application specifically.

Next slide. We've developed a simulator model that incorporates, at the moment, all the thermal-hydraulic aspects of the system. And it's connected directly to the HMI that we are developing for the operators to use. And this will be used for operator training, as well, as we move on. It will also be expanded to include the neutronics processes as well.

The last one. That is our reactor protection system. We are using the HIPS platform that is the license platform by Rock Creek Innovations and we are testing that out as well with all the safety indications. And as I said it's already completely separate from all the other systems.

MEMBER BROWN: Are you using the HIPS system right out of the HIPS topical report?

MR. CILLIERS: At the moment, yes. We're not making changes from the --

MEMBER BROWN: With the volatile/non-volatile processors or FTGAs. Okay.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Yes, this is Jose.

On the topic picture, I see five work stations in the control room. You don't really expect five operators per reactor, do you?

MR. CILLIERS: No, not at all. This is for development purposes. We are most likely going down to three, so the three seats in the front is the operator screens and the two in the back are instructor screens for operation.

MEMBER MARCH-LEUBA: And it looks like a control room for operating a reactor for a light-water reactor, right? I would have expected you to be shooting for one operator at most.

MR. CILLIERS: We are using iterative development. As we learn more, we will implement those type of reductions.

MEMBER MARCH-LEUBA: And that again will be maybe accomplished at the operating license step?

MR. CILLIERS: Yes.

MEMBER MARCH-LEUBA: We have no idea now. Okay.

MR. CILLIERS: Next slide. I think this is the last slide. So this is the indication of our electrical architecture. Our electrical architecture in the PSAR is currently limited to the I level electrical supplies. We don't exactly know all of

those blocks what the exact power demands would be and the voltages required for each different one. Important, yes, is I think the question that came earlier, you will see there is uninterruptable -- uninterrupted power supplies supplied to various systems such as the plant control system, the reactor protection system, the main control room.

All of those have got 72-hour uninterrupted power supply capacity. Should we lose complete power, although we do have normal power supply coming in from a feeder from the utility, as well as backup generators with automatic transfers switching between our normal to backup supply, that automated transfer switch is specified to transfer power within 20 seconds from the normal supply to the backup supply. And for that reason we have a small, short duration capacity to prevent interruption during the transfer, so that the systems that require power do not trip. This is important. They require power to not trip, will not trip during a transfer such as that.

If the transfer fails, that 10 to 20 seconds lapses and then they will trip, so the plant will trip without power if they don't have normal, backup power supply.

MEMBER BROWN: You said a time. You said 20 seconds?

MR. CILLIERS: Yes, that's correct.

MEMBER BROWN: So you're going to have enough built-in capacity in whatever systems, they'll hold up during that open period?

MR. CILLIERS: Yes. It's a very small power supply that's required. They just keep the relays open.

MEMBER BROWN: So you get the backup --

MR. CILLIERS: Until the backup is running, that's correct, yes.

And that's an important point to raise. This is for the safety-related system, but it gives you a better idea of what the architecture looks like to supply the system so that you can monitor for 72 hours after complete loss of power, although in most cases our backup for normal power supply will be available.

I think that was the last slide. Any questions?

MEMBER BROWN: There will be more. This is just an overview, right?

MR. CILLIERS: Of course.

CHAIRMAN PETTI: This is the end of the --

let's just call it the hardware discussion. And then we're going to transition to safety cases?

MR. CILLIERS: Yes.

CHAIRMAN PETTI: Maybe this is the right time to take a break then.

MEMBER BROWN: Good idea.

CHAIRMAN PETTI: So let's come back at 10 after the hour.

(Whereupon, the above-entitled matter went off the record at 3:51 p.m. and resumed at 4:10 p.m.)

CHAIRMAN PETTI: Okay, it's ten after. Kairos, are you ready to continue?

MR. PEEBLES: One second. We're pulling up the slides.

All right, so last up is Jordan with the safety case.

MR. HAGAMAN: All right, good afternoon. My name is Jordan Hagaman. My role is Director of Reliability Engineering at Kairos Power and in the next few slides we'll be discussing the approach to demonstrating margins for nuclear safety for the Hermes construction permit design. I'll explain the approach and the strategy to the design of the safety case, but we're going to rely on other subject matter experts at Kairos to address discipline-specific

questions.

The safety case for the Hermes reactor is described in Chapter 13. First, it's important to understand the overall design of the safety case (audio interference) accident and its relationship to other postulated events.

Coming up, we'll talk about specific events in the safety case and further discuss the MHA analysis itself as well as the approach to analyzing the consequences of postulated events.

The focal point of the safety case is the maximum hypothetical accident analysis which is presented as a bounding demonstration of margins to the dose limits in Part 100 siting criteria.

The MHA is specifically designed to be bounding in a non-physical way which means it's decoupled from many of the specific design details of the future Hermes plant. This should give confidence that the MHA analysis results and conclusions remain consistent over time as the Kairos teams learn from non-nuclear hardware demonstrations and make perfected changes to the Hermes design before it's built.

As we'll discuss, the design of the MHA has built in conservatisms that stress the components of the Hermes' functional containment to overestimate

the postulated release of radionuclide material in the source term.

The maximum hypothetical accident by design bounds the consequences of all postulated events considered for the Hermes safety case. We'll get to the list of postulated events next.

The list of postulated event groups is comprehensive such that the consequences of any postulated initiating event are bounded by the limiting case in one of those groups or the strategy to preclude or prevent that initiator is described and that's also in Chapter 13.

Although the MHA assumptions are largely decoupled from the design features of the to be built Hermes, the postulated event analyses will be more dependent on the final design. Because of this, the preliminary safety analysis at the construction permit stage focuses on qualitative descriptions of how the transients will be bounded by the performance of the plant.

The detailed, quantitative results of safety analysis for the postulated events will provide the final demonstration that the consequences of all of the PEs are bounded by the consequences of the MHA. All of that information will be available in support

of the operating license application itself.

To get confidence at the construction permit stage that the final safety analysis is achievable, acceptance criteria are provided that define figures of merit specific to each postulated event group. These criteria are specifically defined as surrogates that will be used to demonstrate that the bounding case for each event group is bounded by the MHA analysis.

To begin -- take that to the next slide.

As introduced earlier, the MHA is the centerpiece of the Hermes safety case and the MHA is the tool used to quantify margins of safety in the preliminary Safety Analysis Report.

We'll discuss the actual assumptions and margins for the MHA in the next slide. The other seven groups are postulated events that will be demonstrated to be bounded by the MHA. In each of these events, the reactor protection system which we just heard about in the last presentation is available to remove power from key systems to initiate trips of the RCSS, the primary pump, the pebble extraction machine. And passive decay heat removal is available to bring the Hermes reactor to a safe state without recourse to operator actions.

In each event group, the analysis will leverage the high-thermal margins of the TRISO fuel and of the salt coolant to demonstrate mitigation of radionuclide release consequences to lessen those from the MHA.

So I'll describe each event class at a high level and we can come back to it if there's any questions about the strategy for any of these.

CHAIRMAN PETTI: So Jordan, I had a question. If you've seen anything that we've written on the concept of figuring out what events to look at, we talk about starting with a clean sheet of paper and rarely thinking hard about this.

How much of what you did, did you look at other reactor types? Because, you know, this is a unique one. It's neither feast nor foul or whatever that expression is.

For instance, you know, there are a number of fast reactor transients that are out there that one could think about for a system like this. There's a number of pebble bed transients or gas reactor transients that have been done on every small gas reactor that's ever been built in the world.

Did you look at, you know, those sorts of events to come up with a list? Because specifically,

I don't see a lot in the unprotected event category. And again, not arguing, per se, one should say a fully unprotected event, but even a delayed reactivity event.

My perspective here is you guys have an amazingly robust technology because you're marrying two really good technologies, TRISO fuel and salt. And you can be bold in some of these events and show how the inherent features of those two technologies keep the design safe, even in some pretty severe events, more severe than what you've looked at. And yet, you haven't done that.

I just personally think that, from a public safety perspective, that would be a very good position, given you guys have really the first advanced non-LWR to come into the system, that you would be able to demonstrate very robustly with events that even are a little bit more severe than what you've considered.

MR. HAGAMAN: Thank you for that comment. What we're trying to do with the first nuclear demonstration hardware at Kairos is we're trying to develop a safety analysis that can largely bound a lot of the detailed trade space right now which means that we're trying to decouple the safety analysis from a

lot of specific design considerations and further, we're working on a pathway to demonstrate all of our methods. And I believe at this point, it would be too much of a step in one iteration trying to go all the way to, for example, using inherent reactivity feedback to reduce power without shutdown elements.

Right now, we think that it's an easier step to credit reactivity shutdown via shutdown elements rather than inherent reactivity feedback and things like that, so we -- we're not looking to be too aggressive with all of our margin at this point with our very first reactor.

Is that addressing your question?

CHAIRMAN PETTI: I understood that. I was thinking, as I read it, well, maybe you guys plan on doing some transient testing in the reactor, like EBR2 did, or like the small gas reactors did. But then I'm worried that, in a four-year life, you've got an awful lot to do in four years. Those tests take a little bit of time to think about.

Are you still even thinking about those sorts of things because they could so well inform the larger one, you know, you're planning the commercial one when you're planning down the road.

MR. HAGAMAN: So I don't think that I have

more to offer at this time on whether or not we would be considering that.

CHAIRMAN PETTI: Just a member's comment. Something to think about.

MR. HAGAMAN: Okay.

CHAIRMAN PETTI: Because you guys, probably most of you guys weren't even around in those days. I was here when EBI2 was done in Idaho. I mean it was on the front page of the papers. It made a big impact about the safety of the technology and so it's just something to consider.

MR. HAGAMAN: That's a good comment, but at this point we don't have anything in chapter --

(Simultaneous speaking.)

CHAIRMAN PETTI: Right, I saw that.

MEMBER REMPE: So this is Joy, and I have a question that is just general. Could you explain to us how you track all the assumptions for, which data don't yet exist, that you used in these analyses? And is there some system that the staff can audit that shows all of these assumptions? And if that list exists, you also have a comment after then that says we're planning to get this data in such and such a facility?

MR. PEEBLES: So, just at a high level,

to answer your question, this is Drew Peebles. We do have something that is available to the staff to audit. And there is kind of an audit open on Chapter 13. I won't get ahead of the staff on their review, but yes, the answer to your question is that information is available to the staff.

MEMBER REMPE: And so just generally has the staff been -- maybe they haven't finished the review enough, but if a list does exist so they can audit it, do they sometimes in some of the interactions that are ongoing they've identified additional assumptions that -- I mean, is this back and forth yet? Or you haven't gotten that far in the process with the RAIs?

MR. PEEBLES: Yeah, I think I better answer those questions after the review is completed. We're still in audit discussions right now.

MEMBER REMPE: Okay. Thank you.

MR. HAGAMAN: So I was prepared to just speak briefly about what is grouped in each of these postulated event groups for awareness. As a reminder, these are all of the groups that we are, by design, are making sure that our maximum hypothetical accident analysis bounds.

So where the MHA is our tool to

demonstrate margins, these other events, we come up with other methods to show that they are bounded by the MHA and therefore have at least the margin that the MHA demonstrates.

So the first group is insertion of access reactivity. This is a group of events that includes reactivity and insertion of events ranging from fuel loading errors to increase in heat removal events and overcooling to phenomena associated with shifting reactor blocks or movement of gas bubbles.

The salt spill events involve a loss of primary coolant from the primary heat transport system. I want to note here that the preliminary Safety Analysis Report deliberately describes events as salt spills and not loss of coolant accidents. We do this to avoid confusing the phenomenology of light-water reactor LOCAs with the phenomenology of the Hermes reactor where coolant spills have significantly less safety significance.

MEMBER MARCH-LEUBA: Let's just stop there. Let's stop there. A salt spill is not called a LOCA. What happens if you spill enough salt that you uncover the return path so that your natural circulation doesn't work anymore. Again, a LOCA of the return line that goes to the feedwater, I mean the

one that goes to the heat exchanger, if that line breaks down, everything drains through it, not only drain through it, you stop the natural circulation and you have a path out to the vessel for all the --

MR. HAGAMAN: I understand. The salt spill, we look at the entire spectrum of locations and sizes for leaks including on the cold leg and we have features built into the vessel to break the siphon should a leak in the cold leg happen. So the siphon breaking --

MEMBER MARCH-LEUBA: How about breaking the hot leg?

MR. HAGAMAN: I'm sorry, can you repeat the question?

MEMBER MARCH-LEUBA: How about breaking the hot leg?

MR. HAGAMAN: Hot leg as well. The features are actually built into -- I want to make sure I get the terminology correctly, the pump casing itself, I believe, to break the siphon.

MEMBER MARCH-LEUBA: So the hot leg is pumped from the pump from the top of the vessel?

MR. HAGAMAN: Correct.

MEMBER MARCH-LEUBA: What I'm looking for is a possibility of draining the vessel, having a salt

spill, either through a LOCA or through an inadvertent action of something that will lower the level below and not the circulation.

MR. HAGAMAN: So I have a team of people that is also looking for that scenario and we are looking for the list of assumptions where we can say that that is precluded. So we've done a lot of work in that area. The anti-siphon features, limitations on gas entrainment, as well as the trip timing for the pump are all -- and the elevation of penetrations are all part of the series of design characteristics that we're going to be relying on to preclude that event.

MEMBER MARCH-LEUBA: So they will -- you promised, I think you promised there will be design features that will prevent this, but they don't exist now?

MR. HAGAMAN: So Chapter 13 describes what we call the list of prevented events. This is part of the uncooled event where, for whatever reason, we look for all of the ways that we could lose capability to remove to decay through our DHRS system. And that is one of the areas where we're identifying the design features. We've done that at a high level in the preliminary Safety Analysis Report. And what you can expect is as part of the application for an operating

license, we will have all of the features and controls that include both design features and programmatic operational features to ensure that this event stays precluded through the life of the plant.

MR. PEEBLES: This is Drew Peebles. I have to add that this isn't a pinky promise that this is going to happen at the OLA (phonetic). We have hard commitments in the PSAR that we will maintain the Flibe level above the active core for both normal operations and all postulated events. But there is no spill that we've identified that could drain the Flibe below the top level of the active core.

MR. GARDNER: This is Darrell Gardner. I would also add that currently in the PSAR, for any penetration into the vessel, we have already describe how we address, functionally, any seismic scenarios. So if you go and look, for example, Chapter 9 for systems that connect to the vessel, or if you look in Chapter 5, which is the inlet and outlet PHCS lines, that's already discussed.

MEMBER MARCH-LEUBA: I'm looking forward to looking through all of that. I'm looking at Chapter 15 -- 13 and it's very light on details.

MR. PEEBLES: So those are event analyses, but again, in the system design you will see the

discussions of the design features to preclude drain down events.

MEMBER MARCH-LEUBA: I don't see anywhere saying -- I mean you always say keeping the core covered. What you need to say is keeping the natural circulation path covered.

MR. PEEBLES: That's also in Chapter 4. Chapter 4 has commitments for both natural circulation and keeping the core covered.

MEMBER MARCH-LEUBA: We'll look at it in more detail when we review chapter by chapter. Right now, I don't have much hard feeling that everything is -- I think if you start with assumptions that everything is super safe and work backwards, that's the impression I'm getting.

MEMBER REMPE: It does seem like the welds holding the bottom plate of the vessel to the cylinder are going to be very robust to never fail.

MR. PEEBLES: Yes, but that's not dissimilar than the light-water reactor vessels.

MEMBER REMPE: There are hemispheres and they're welded to different locations other than a flat plate. I'm just kind of thinking about it. I don't have an opinion yet. I'm just exploring it.

CHAIRMAN PETTI: Keep on going, Jordan.

MR. HAGAMAN: All right, so as discussed, salt spills include the spectrum of leak sizes and locations throughout the non-safety-related primary coolant system where we leave off. The loss of forced circulation is an event group that includes a range of events from mechanical or electronic failures of the primary pump during operation to flow blockages in the primary coolant system, all the way to just normal loss of -- normal heat sink events or even a loss of power. Those are all grouped under loss of forced circulation.

The mishandling or malfunction of pebble handling in storage systems, this is a group of events that includes pebble transfer line breaks for lines that bring the pebbles into the empty core or the at-power core, all the way to the lines transferring pebbles to storage containers. So we look at the potential for malfunctions or breaks in all of the lines there.

The radioactive release from a subsystem or component. This is a standard category from NUREG-1537 for an FHR. For Hermes, it includes faults in the tritium management system, the inert gas system, chemistry control, inventory management.

General challenges to normal operation

includes spurious control system trips, inadvertent operator actions. There's a suite of possible events that you could expect inside of what Anthony in the last presentation showed as his blue box. All of these events we expect to be bounded by the worst event and the loss of forced circulation.

And the internal and external hazard events, this largely goes back to Chapter 2 of the preliminary Safety Analysis Report in Brian Song's presentation from earlier today where internal fire, internal and external floods, seismic, high winds are all evaluated against their potential to interrupt the function of safety-related SSEs. We build that into the design basis.

So we're ready to jump into the slide about the maximum hypothetical accident. So in order to demonstrate in the construction permit application that the Hermes maximum hypothetical accident is, in fact, a sufficiently conservative hypothetical event, the PSAR points out specific, non-physical assumptions that are meant to challenge the elements of functional containment, namely, that's to drive the fusion of radionuclides through TRISO layers and to increase the evaporation of radionuclides from the free surface of the Flibe coolant itself.

The MHA analysis presented in Chapter 13 heavily leverages the methodology for -- that was in the source term topical report that was under ACRS review last year. That's KP-TR-012. The specific assumptions include pre-transient diffusion of radionuclides as neglected. This takes a little bit of explanation.

This assumption maximizes the amount of material at risk which accounts for inside the TRISO fuel itself. By neglecting the fact that during normal operations, material will naturally transport and diffuse through TRISO barriers in steady state before a transient condition which would deplete that source of material at risk. We neglect that phenomenon to maximize the amount of material at risk in the fuel.

But at the same time, the amount of material at risk assumes in the salt itself, reflects an upper bound of the opposite assumption where a maximal amount of material diffuses in steady state from the fuel to the salt. So we have a hypothetical super position of these two assumptions that effectively double counts for the material that would move from the fuel to the salt during normal operation. So for the accident analysis, rather than

making a choice between in the fuel or in the salt, that material is both in the fuel and in the salt for the purposes of a hypothetical accident.

Hypothetical temperature histories are presented in Chapter 13. These are specifically designed to drive radionuclide release from the TRISO fuel where diffusion happens at a higher rate at higher temperature to drive radionuclides from the graphite structures and to drive radionuclides from the Flibe salt coolant where evaporation of different species is higher with higher temperatures.

These are artificial, prescribed, flat temperature profiles and while they drive the release of radionuclides and the MHA, they also are important input to the definition of figures of merit for the postulated event analyses.

The next assumption has to do with the gas base itself. And essentially everything that leaves the free surface of the Flibe coolant is free to transport to the site boundary in analysis space with minimal reliance on the confinement of radionuclides within the gas boundary itself and within the reactor building itself. And so we minimize reliance on retention in any physical structures and we -- in the analysis presented in Chapter 13 are mostly focused on

the mitigative capability of both the TRISO fuel and the salt.

Lastly, there's tritium modeling. We take a conservative approach in two ways. First, the tritium content that is inside the salt coolant and inside the carbon matrix of the pebbles is assumed to hypothetically puff release at the beginning of the transient. That's a very non-mechanistic, hypothetical way to treat that material at risk.

The second way is the tritium content within the graphite reflector structure itself is released by a bounding diffusion model that's driven by a time and temperature curve which is also in Chapter 13.

So the table at the bottom of the slide shows the demonstration of margins of safety for the Hermes reactor that's presented in Chapter 13. Our criteria is the siting criteria in Part 100. That includes the limit on whole body dose at the boundary which is 25 rem for the worse two hours of the accident. And also, thyroid dose has a limit of 300 rem.

Against these limits, the Hermes safety analysis demonstrates over 24 rem margin to the whole body limit and over 299 rem to the thyroid dose limit.

Before I move on to the next slide, are there questions?

CHAIRMAN PETTI: Just a quick question. You guys are the first ones really employing functional containment and there's not a lot of details here in the PSAR, you know. There's a jump from the topical report on source term to this. We will be identifying cross-cutting issues and although your numbers are low and I sort of don't -- I think based on what you're saying they kind of make sense to me.

Will there be a document that we could look at to look at the release faction from the TRISO or the release faction from the salt, something that puts the pieces together?

I just think -- not from a standpoint of did you do it right, but because you're the first using the functional containment, having some of that data out there and having the ACRS being able to make a statement about that I think is important. And it's something that probably would only have to come through in an RAI would be my guess. So it's also a note to the staff.

I think, you know, given you guys are the first, there might be some value there.

MR. HAGAMAN: I appreciate the comment. The specific details are not in the Preliminary Safety Analysis Report. I don't want to get too far ahead of the NRC staff's review of what they think is adequate to substantiate it, but I do appreciate the comment and I do recognize that this is the first application to use the functional containment and those details are --- they certainly are of interest to us internally and of interest on the on-going NRC review and I think that's as far as I can take it right now.

CHAIRMAN PETTI: No, that's fine. It's out there. Thanks.

MR. HAGAMAN: All right, so the last slide, very briefly, as we discussed already, the methodology and the sample results of the postulated event analysis were provided with the expectation again that the real quantitative results, based on a more final version of the design will be available in support of the operating license application.

The postulated event methods are provided in the report KP-TR-018. You'll also see in there some sample calculations that illustrate how the methods will work to demonstrate that the events are bounded by the consequences of the maximum hypothetical accident analysis.

For each postulated event group, acceptance criteria are defined and reported in both Chapter 13 and in that postulated event technical report. The acceptance criteria defined for the figures of merit and these criteria will ensure that the limiting case in each group has consequences that are bounded by the MHA where we're demonstrating our dose margins, meaning that the safety case relies on surrogate criteria rather than full dose consequence analysis for each minor event in the safety case.

As stated earlier, validation of the models and the detailed final analyses of the specific postulated event groups will be available in support of the operating license application.

Thank you for your time. I look forward to questions now and during the review later this year.

MEMBER REMPE: Dave, this is Joy. I have a question for you. I got assigned, I believe, the event analysis review and Walt's assigned to the Chapter 13 review and I think these two reviews are very closely related.

Can we plan to our discussion on them at the same time when we go through the future assignments or schedule?

CHAIRMAN PETTI: So 13 and which one?

MEMBER REMPE: This KP-TR-018.

CHAIRMAN PETTI: Oh, right, right, right, right. Yes, yes, yes.

MEMBER REMPE: And I don't know if the staff is doing a SC on the topic report or the technical -- but I don't think they usually do. But let's do them together.

CHAIRMAN PETTI: Yes, no, I think there's a lot of things where the order in which we do things will be important, so we need to work with the staff on that.

MEMBER REMPE: Okay. Thank you.

CHAIRMAN PETTI: So we don't drag it out. Yes. I see Vicki has her hand up.

MEMBER BIER: Yeah. Vicki Bier. Quick comment. This is really just following up on the discussion with Jose earlier about the detailed design features that preclude various types of events.

I don't want to make too big a deal out of the wording, but the language that was used is, we are looking for assumptions that allow us to preclude. And really, the thought process should be, we are looking for possible paths by which this could happen, by which you could get in trouble, and then preclude

those paths.

And I don't think it's what you're doing, but there's a risk that if you're trying too hard to prove what you want to be true, you may prove it even though it's not true. So just a caution on that, but not a big concern at this point.

MEMBER MARCH-LEUBA: Yeah. And this is Jose. This is what we've been calling start with a blank sheet, and that always means a blank sheet is having a questioning attitude: what can possibly go wrong? And it's not clear that we are having that, to me.

So, certainly, with the detail available -- for example, this anti-siphoning (phonetic) thing that we've been taking credit for, there's a circular logic in the PSAR. Chapter 5.3 goes to 4.3, who goes to 12.2, who goes to -- eventually back to 4.3 and another describes. Yeah. There will be lots of questions when we go over chapter by chapter.

MR. HAGAMAN: Thank you for the comments.

CHAIRMAN PETTI: Okay. So, just before we -- you guys will be done, right? We'll be transitioning to the staff next, this --

MR. HAGAMAN: Yes.

CHAIRMAN PETTI: Yeah. I just want to put

on the record that I particularly like your technology development path, that slide, your testing, testing, testing. I think that's -- it's going to inform the design tremendously and fill in a lot of gaps. I mean, some of the stuff, not until you get to Hermes will you really know. But for a facility that's never been built, for a technology that's never been built before, I think it's commendable the approach you guys are using.

With that, let's get the staff up because we are really running out of time. My guess is that we'll be done before 5:30, hopefully, but we're probably going to go over a little bit.

MR. HELVENSTON: Yeah. This is Ed Helvenston with the staff. I'm trying to get the slides loaded up now. Are they showing up okay?

CHAIRMAN PETTI: Yep.

MR. HELVENSTON: Okay. Perfect. I'll go ahead and get started, then. My name is Ed Helvenston. I'm from the Non-Power Production and Utilization Facility Licensing Branch in the Division of Advanced Reactors and Non-Power Production Utilization Facilities in the NRC's Office of Nuclear Reactor Regulation.

I'm one of the three project managers for

the staff's safety review of Kairos's construction permit application. And I'd like to follow the technical presentations we just heard with a staff presentation in which I'll give a brief overview of the staff's review process and schedule, and I'll also try to touch on a few other important topics relevant to the staff's review of a construction permit application for a non-power testing facility.

So, as you know, the NRC's receipt and review of applications for construction and operation of new reactors based on novel technologies, such as Hermes, is an important milestone in the success of advanced nuclear technologies in the U.S. Although it's the responsibility of Kairos as the Applicant, and other designers, to demonstrate the safety of their designs, the NRC staff must perform its mission of independently reviewing the safety of these designs in an efficient and effective manner.

Accordingly, the staff's review of a design such as Hermes will be focused on the matters that are most safety significant. The scope of the staff's review of a design is commensurate with the risk posed by a design. Performing an efficient and effective review of a design such as Hermes warrants innovative and novel approaches.

So, in terms of overall responsibility for the staff review of the Hermes application, this lies with the Division of Advanced Reactors and Non-Power Production Utilization Facilities, also known as DANU, which is in the NRC's Office of Nuclear Reactor Regulation. So DANU has primary responsibility for licensing activities for all 10 CFR Part 50 testing facilities, including non-power testing facilities using advanced technologies such as Hermes.

One example of an innovative and novel approach that the staff is using for the Hermes review is the staff's core team approach. To support an efficient and effective review, what the staff has done is we've assembled a core review team of near full-time and significant part-time staff, which includes two advanced reactors project managers from DANU, a non-power reactor project manager from DANU -- myself -- technical reviewers from DANU, as well as an attorney from OGC.

In lieu of divvying specific review areas among a wider array of technical reviewers as we've done with many reviews in the past, in the core team approach, we have DANU technical reviewers with significant advanced reactor technology expertise who are taking responsibility for broader portions of the

application as well as gaining an understanding of the overall design.

The types of technical topics that are being reviewed by the DANU core team include many of the topics that are integral to the reactor design, such as thermal and structural analysis, fuel and core design, and accidents. Some of the other types of topics that are being reviewed outside the core team, similar to a more traditional approach by subject-matter experts, include areas such as, for example, quality assurance, fire protection, site characteristics, and emergency planning.

CHAIRMAN PETTI: Ed, just a question.

MR. HELVENSTON: Yeah.

CHAIRMAN PETTI: Is reactor physics sort of a subset of one of those things you had on -- items on the previous slide? Is it in, like, fuels or in --

(Simultaneous speaking.)

MR. HELVENSTON: I'm not sure that it really fits into any of the ones here. These are just example topics. I wouldn't call this an exhaustive list.

CHAIRMAN PETTI: Okay. Just with a moving fuel system, again, it's the first time the staff has seen it. I think we want somebody that understands

physics and shut-down margins and all those physics-y things.

MR. HELVENSTON: Absolutely. We're certainly looking at those areas as part of our review.

CHAIRMAN PETTI: Great. Thanks.

MR. HELVENSTON: So, in terms of NRC licensing of non-power reactors, a 10 CFR Part 50 license for a non-power reactor could either be issued as a Class 103 license for a commercial facility or a Class 104(c) license for a research and development facility.

In accordance with the NRC regulations in the Atomic Energy Act as amended, any Class 104(c) facility must be useful in the conduct of research and development activities of certain types that are specified in Section 31 of the Atomic Energy Act. The specific distinctions between 103 and 104(c) are based on certain financial tests that are described in NRC regulations in the AEA about how much the cost of operating a facility is spent on and recovered from commercial activities as opposed to R&D activities.

And in its construction permit application, Kairos has stated it plans to apply for a Class 104(c) utilization facility operating license.

And accordingly, the staff is conducting its review of the Hermes C application consistent with the requirement that's given in Section 104(c) of the AEA that in order to permit the conduct of widespread and diverse research and development, the Commission imposed only the minimum amount of regulation needed to permit it to fulfill its obligations to promote common defense and security and protect health and safety.

So types of non-power reactors that are defined in NRC regulations include both research reactors and testing facilities. A testing facility as defined in Part 50 is a reactor designed to operate at a thermal power in excess of 10 megawatts or in excess of 1 megawatt if the reactor is to contain certain features.

A research reactor is, in general, a non-power reactor that is not a testing facility, for example, because its thermal power is below 10 megawatts. Per the Part 50 definition, a testing facility may also be a reactor of the type described in 10 CFR 50 21(c), in other words, a Class 104(c) facility that is useful in the conduct of research and development.

Many of the prescriptive requirements in

10 CFR Part 50 are only applicable to nuclear power reactors and therefore do not apply to non-power research reactors and testing facilities. However, testing facilities are subject to the siting requirements, including accident reference doses in 10 CFR Part 100.

Testing facilities are also subject to a few 10 CFR Part 50 requirements that do not apply to research reactors, including a requirement for ACRS review of CP and operating license applications, as well as mandatory Commission hearings for CP applications.

This slide just gives a brief overview of the licensing process for a testing facility such as Hermes. The process is generally similar during the CP and OL application reviews. When an application is received, the staff first performs an acceptance review of the application to determine whether the application contains sufficient information in scope and depth for the staff to begin its detailed technical review of the application.

Once an application is accepted, the staff begins its separate safety and environmental reviews in parallel. For the CP, the product of these reviews is a safety evaluation report, or SER, and

environmental impact statement, or EIS. And for the OL, another SER and EIS supplement are prepared.

Once the staff completes its safety review and SER for a CP or OL, ACRS meetings are held, and following the ACRS review and issuance of the ACRS letter to the Commission, a Commission or Atomic Safety and Licensing Board, if delegated by the Commission, hearings are held on the Commission as applicable.

For a CP, a mandatory hearing required by 10 CFR 50.58 is held on the sufficiency of the staff's safety and environmental reviews for issuance of a CP. In addition, for either a CP or OL, there is a potential for separate contested hearings on the staff's safety or environmental reviews if requested by interveners. Following any hearing or hearings, a decision is made to grant or deny a permit or license.

So, as consistent with the minimum regulation requirement that I mentioned in Section 104(c) of the Atomic Energy Act, and also consistent with the need to perform an efficient and effective review of the Hermes CP application, the staff will perform a risk-informed review in that its review depth and scope will be commensurate with the safety significance of areas under review in the application.

The staff is maintaining a big-picture safety perspective of the Hermes design and is tailoring the scope and level of detail review based not only on the small size of Hermes but also on the anticipated strong safety case and low radiological consequences and considering that the application is a CP application for a testing facility.

The staff is also tailoring its review to the unique and novel Hermes technology described in the CP application. The staff is using NUREG-1537, which is the licensing guidance for non-power reactors, in performing its review. NUREG-1537 is designed to be technology neutral and provides flexibility for a review such as the Hermes review. In addition, NUREG-1537 Part 1, which provides guidance to applicants, is the guidance that Kairos used in preparing its CP application.

MEMBER KIRCHNER: Edward, this is Walt Kirchner. Just a rhetorical question on your previous slide, and you had mentioned this earlier, that you look at the risk posed by a new applicant. So this comment is independent of Hermes. In general, how do you assess that risk going in?

What I mean is you have an advanced reactor concept that hasn't been built and operated

previously. The risk is two things. One is the frequency of potential for something to go wrong, and the second part is the consequences. So, when you talk risk in this sense, it seems to me you're using that as a surrogate for the source term, essentially, that the reactor has -- in other words, the thermal power. Is that how you look at risk, or how do you make that screening decision?

MR. HELVENSTON: The thermal power is certainly one factor that you look at, but I don't think we'd want to limit it to thermal power. Other things would be the technology that they're using, and in the case of Hermes, we're looking at the functional containment concept and the idea that the Flibe and the fuel will be able to retain fission products.

And it's a combination of the factors that you're looking at, really, in determining the overall risk and scaling the review of specific areas of the facility appropriately with that. Yeah. I think you would consider a wide range of factors in scaling your review.

I'd say the staff would start with -- we can start with an assumption of low risk, and we can work with that, but certainly as we do our review, that's something that we need to verify. And if we

find something that changes that assumption, then maybe that's something that we need to go back and take a closer look at.

MEMBER KIRCHNER: Thank you.

CHAIRMAN PETTI: Vesna, you have a question?

MEMBER DIMITRIJEVIC: Yes. I have a question related to this because I'm always ready -- when we say risk informed, we should really define what risk we are talking about because if it's risk informed, then it has to base on some metrics, right? So, now, in your answer you gave so many of the general -- you know, thermal energy, thermal power, the containment.

And we were expecting there will be dose related. So when you're doing this review, how are you looking at this? Do you say, for example, that your main metrics is, for example, vessel integrity and foil integrity, or your main metrics is dose, or your main metrics is challenges to this, or you are -- or maybe when you're doing review, you're just looking in the safety systems and accident analysis.

How are you positioning yourself in this review? What is your metrics when you're talking safety this -- because there is no risk measures and

there is no risk relative ranking of the systems. So what is your risk metrics?

MR. HELVENSTON: Well, I think at the end of the day, the risk metrics we're really looking at is health and safety. And that's quantified through what the prospective dose could be. But we have to look at a number of things kind of intermediately to get there. We're certainly looking at the safety systems that -- any mitigation functions they have and the technology and how it's designed for mitigation.

MEMBER DIMITRIJEVIC: Okay. So it would -- all right. So you're basically looking in the mitigation of the maximum accident or all other accidents which are considered -- okay. That will be interesting to follow when we go through the review; how did you position yourself in this prioritization of the review? Okay. Well, something to think about. Thanks.

MEMBER REMPE: So I have a question, and I'm afraid I'm going to misquote what the Applicant said. But they said they didn't ask for a finding on safety at this time; they wanted a finding with respect to how much margin would be required by the staff.

And could you elaborate what, if -- I'm

probably misquoting exactly what they said, but how you plan to do that? For example, if it were on the dose to the 10 CFR 100 limits, are you going to say, okay, you got to have at least a factor of 100 because there's so much uncertainty in the data? Or what is it they're asking for, and then how are you going to plan to get there is what I'm curious.

MR. HELVENSTON: I don't want to speak for Kairos, but they haven't requested final approval of any portion of their design in this application. So we're not approving a final design as any part of our CP review, at least with what's been requested at this point.

I do have a slide a couple slides from now -- I will talk a little bit about the findings that we are looking to make for a construction permit and kind of how we determine what we need to look at in the CP versus what we reasonably believe can be put off till the OL, if that might be helpful.

MEMBER REMPE: I took a peek at that, but I didn't see what -- I thought I heard them saying -- maybe they can speak up and clarify if I'm misquoting them, but I thought that the guy that was the head of the licensing came in when Jose was asking questions, and he said, oh, we're not asking for a finding of

approval of the design with respect to safety. We did ask for something with respect to how much margin the staff would want.

And I was real curious because I wasn't sure of that. But you've not done anything in your last slide with respect to how much margin you're expecting them to come in with that I can see.

MR. HELVENSTON: Yeah. I don't remember the specific comment from Kairos. See, and if anyone from Kairos wants to speak up and clarify, that's fine.

MEMBER REMPE: Yeah. They asked for three things, they said. And so, yeah, I'd like to hear those again very carefully because Jose said he was going to put it in the first paragraph of our letter.

MEMBER MARCH-LEUBA: Well, it may be the last paragraph in our comment. But if you don't know what they are asking to review, that's not a very good statement to say in the --

MEMBER REMPE: No. Let's ask the Kairos to clarify again, what were the three things they asked for?

MR. GARDNER: Sure. This is Darrell Gardner again from Kairos. I think what --

(Simultaneous speaking.)

CHAIRMAN PETTI: We can't hear you, Darrell.

MEMBER REMPE: Darrell, could you speak up again, please?

MR. GARDNER: There we go. How about now? Can you hear me?

CHAIRMAN PETTI: Yeah. Now we can.

MEMBER REMPE: Much better. Thank you.

MR. GARDNER: Okay. So what I was saying was somewhat paraphrasing from the findings that are required in 10 CFR 50 35, so what the Commission is required to conclude in 50 35 and what the requirements are in 50 34(a) for a preliminary safety analysis report. And so those are different from -- there not findings of final safety.

In fact, the language in 50 35 is fairly clear that it doesn't represent any findings of final safety acceptance of the design unless the Applicant specifically requests for that, which we have not. But it's simply an authorization to proceed with construction. The exact language is the authorization to proceed with construction but will not constitute Commission approval of the safety of any design feature or any specification, less the Applicant specifically requests such approval.

So the point I was trying to make was if you go back and look at the requirements of 50 35(a), they are not demonstrating final safety design. What it asks for is margins to safety, which we believe we're providing with the MHA analysis.

MEMBER REMPE: So you're not asking for them to say how much margin to have; you just want them to determine that there is sufficient margin in some vague sense? Because I wasn't sure of what I heard, but I might have misheard you. Am I better saying --

MR. GARDNER: I would say that that's correct, absent the word vague. But yes.

MEMBER REMPE: Okay. Got it.

MR. HELVENSTON: Yeah. I'll just add that I'd want to look at the wording of the regulation, but in terms of margin, as Darrell said, Kairos is not requesting and we're not making a final safety determination.

But the margins are certainly one thing that the staff would look at in the CP that would support its -- making sure there is reasonable assurance of adequate margin to support our conclusions that we need to make for the CP, including that there is reasonable assurance that the final

design is going to conform to the design bases.

MEMBER MARCH-LEUBA: Yeah. But how can you be even confident -- not sure, generally sure, but confident -- that you have evaluated the risk and that you can say that this design has margin to safety if you have not performed an evaluation? The Achilles heel of all of this research analysis, I keep saying, is completeness.

If you forget in your analysis the most limiting event that actually melts the core and produces a 10 CFR 100 dose, but you didn't analyze it, then we keep saying there is no risk, I have a lot of margin, but you didn't do the analysis. So, if you don't do a thorough analysis, I don't think you can say in your SER that we believe there is plenty of margin to safety. You can say it's possible, but we haven't done a complete analysis. You never do a complete analysis.

So, I don't know. It is a bad position to be on, but you should be very clear on the SER that you have not performed a full evaluation, because, honestly, I see an Olympic lack of rigor, especially on the part of the staff, when it comes to an analysis of safety.

And on the Applicant, I suspect the

Applicant has a lot more background documents that they're not showing, and more background thinking that they're not showing us, that probably supports the position. But, on the part of the staff, I don't see it.

MEMBER REMPE: Jose, how can you say that when you haven't seen the SE from the staff?

MEMBER MARCH-LEUBA: I don't see an attitude if I'm going to do a full safety --

MEMBER REMPE: I haven't seen that yet. All I've seen is some slides about what they're going to do. So I'm going to mention the previous reviews, but we've come up with --

(Simultaneous speaking.)

MEMBER REMPE: Okay. So you're talking about something else, but it's not here. Okay.

MEMBER MARCH-LEUBA: It happens. It happens. It happens, and if you do a full analysis and rigorous and really starting with a white piece of paper, you always find something else. So unless you have a rigorous approach or you have -- honestly, what can possibly go wrong, instead of how well the we fix what we thought of -- I don't know. I'll leave it there.

MR. HELVENSTON: Yeah. Well -- and the

staff will certainly consider that feedback. But we're still in the process of our review right now. We certainly have a lot of questions in these areas as well. I think it's been mentioned we have an ongoing audit in terms of accident analyses right now, and we're looking at some additional information that goes in depth that supports some of what's in the PSAR.

So I'll just say the staff is certainly looking at that and using its technical judgment to make sure we ensure that the accident analyses are comprehensive.

MEMBER MARCH-LEUBA: Yeah. I checked if more information was on the -- access to more information than was on the PSAR because I haven't read it in detail yet because we're not performing the review yet. But as I said, I've been looking through his work while we were talking, and it's a circular logic. It goes from Chapter 4.3 to Chapter 13-point-something to Chapter 12, and nowhere anything is defined.

So I'll be looking forward to see if there is any meat to the conclusions. And as I said -- I mean, let me just -- want to put it in the record. Number one, I think the Kairos and Hermes design are cool. They are cool. They're really safe reactors.

They use the best technology available, all that. I mean, this should be an example for the whole -- but that doesn't give you carte blanche to not do the job.

Again, I think the Applicant is doing the job, but in their offices. They're just not putting it in the paper. So you have to show me you have rigor on your what-can-possibly-go-wrong analysis. And one more thing --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Yeah. My house is located 15 miles downwind from the location of this reactor. And so I have a conflict of interest making sure that this thing works. And even with my house 15 miles downwind of the reactor, I think it's a cool reactor. I want it built. But I want it built safety.

CHAIRMAN PETTI: Dennis, you've been waiting.

DR. BLEY: Yeah. Yeah. I've been just sitting here. When this whole thing came up during the Applicant's discussion, I was a little taken aback by the -- it's become clear it wasn't meant this way, but by the idea that staff shouldn't be looking at safety in a construction permit. And that's not right.

And you are, again, looking at margins without looking at safety. But at least from my point of view, to get through and get a construction permit, you need to show that you've looked pretty hard for, as Jose says, the things that can go wrong, the accidents that can happen, and that there are plausible ways to deal with that.

And you don't do all the analysis. You don't make a final safety finding at this stage. But you have to make sure there's nothing hiding there that implies there's a high chance that you'll never be able to build this thing or that when you build the structures during construction, you aren't locking yourself into an area that could lead to high risk later on. But I think you're doing that. So that's all for me.

MR. HELVENSTON: Thank you for that feedback.

CHAIRMAN PETTI: Keep going, Ed.

MR. HELVENSTON: All right. Well, on this slide, I don't have much to say. I won't talk to this in much detail. But this is just a list of the chapters in NUREG-1537, which is similar to the layout in Kairos's PSAR for Hermes, and we'll also use this as the basic, basic format for the staff's SER.

And also, as noted on the slide, there's a few chapters on here, for example, 15, -- 16, 17, and 18, rather, that will not be applicable to this review.

MEMBER REMPE: This is Joy, and I just kind of wanted to bring up one thing that I thought was important in some of the prior CP reviews. There has been a section, an appendix or something, that lists all of the assumptions and areas where further work is needed. So it just is a nice way to keep track of everything that -- the gaps. And will staff produce such a list for this review? Have you guys made a decision on that?

MR. HELVENSTON: We haven't made a decision on that. That's certainly something we are considering if there is a need to make sure we have those types of commitments documented in one place. I'm aware that we've done something similar with a couple of the NPUF CP reviews in the past.

MEMBER REMPE: And one member's opinion, I think it's a good idea to do something like that. And it might be longer in some designs than others.

MR. HELVENSTON: I think I see one other hand raised.

CHAIRMAN PETTI: Dennis, did you take --

MEMBER MARCH-LEUBA: That's Dennis.

CHAIRMAN PETTI: Did you take your hand down? Thank you.

Jordan, did you have something you wanted to add?

MR. HAGAMAN: No. That was from earlier. Thank you.

CHAIRMAN PETTI: Okay. Thanks.

Keep going, Ed.

MR. HELVENSTON: I mentioned a couple slides back that I wanted to talk a little about -- one other area I wanted to highlight was the consideration, as we've discussed quite a bit already, that Kairos has submitted an application for a Hermes construction permit, and the staff is conducting its review accordingly.

So safety reviews for either a CP or an OL application are conducted in accordance with NRC regulations. For CP, the level of detail in an application and associated NRC staff review are different than what is needed for an OL or a combined operating license. A CP application describes a preliminary design of a facility, while an OL application needs to describe a final design as well as additional administrative plans and programs that

are not provided in the CP application.

So the guidance in NUREG-1537 does not differentiate between the level of detail that is needed for a CP versus an OL application, nor does it provide specific guidance on what types of things may be deferred to the OL. However, in making this determination on types of things that may reasonably be deferred versus what is required for a CP, the staff is using its technical judgment, and we also certainly consider the requirements in 10 CFR 50 24(a) and (b), which regard information that must be included in either both preliminary and final safety evaluation or safety evaluation reports -- or, I'm sorry, safety analysis reports.

In addition, the staff bases its review on the specific findings it needs to make for the issuance of a CP, which are given in 10 CFR 50 35 and I have listed here. So, as provided by 50 35, the principal architectural and engineering criteria for a design must be described in the CP application, but some technical or design information may be left for later consideration in an OL application.

And in addition, not all safety questions need to be resolved for the issuance of a CP, but an applicant must identify research and development which

has to be completed prior to the completion of construction to resolve these questions. In addition, in making a recommendation that a CP should be issued, the staff also considers the requirements in 10 CFR 50 40 and 50 50.

So I think this is my last slide. This is just giving the staff's current schedule for the Hermes review. Given the extensive pre-application engagement for the Hermes CP review, the staff was able to establish an aggressive review schedule of 21 months from application acceptance, which includes ACRS review but does not include a mandatory Commission or Atomic Safety and Licensing Board hearing.

The staff accepted the application for review in November 2021 and completed a draft SER with open items last month. The staff is currently conducting audits in a variety of areas and also preparing additional audits and possible requests for additional information to support closure of open items and completion of all SER chapters by November.

The staff also plans to complete management and OGC reviews and approvals of the SER by May 2023. I will note that the November 2022 and May 2023 dates on this slide are for completion of all

chapters in the SER. The staff intends to complete some individual chapters ahead of these dates as possible and plans to share complete chapters with the ACRS well ahead of the May 2023 day as it is able.

To support completion of the review within the 21-month schedule, the staff is targeting September 2023 for issuance of an ACRS letter for Hermes.

MEMBER REMPE: So this is Joy, and I was thinking about this slide a bit more. And I'm wondering -- we are involved in a review of an OL for another NPUF, and it came to us with all of the open items resolved. Sometimes we lost some of the development activities because the staff didn't include how -- we ask all of these REIs, and this is how the issues were resolved.

And in light of that, we actually ask -- we see the earlier draft SE, and we actually also -- with this accelerated schedule, I'm thinking it might behoove us to actually -- for those members who have time and have other commitments later on the line might want to see the draft SE too.

So have you guys made the draft SE that you've got available to us? I know it won't be something that we'll be discussing ever in an open

session what's in it, but it would behoove us to kind of have access to it earlier in some cases --

(Simultaneous speaking.)

MR. HELVENSTON: We didn't have any specific plan for that. I think that's something that we have to discuss among the staff. I'd say certainly, as you might be aware, there's initiatives in terms of streamlining SEs and kind of focusing them on really the most safety-relevant information, and not necessarily including the level of back and forth on RAIs and that type of thing, since that information is already documented on the docket somewhere else.

But that's something that maybe when we get a little bit closer to the November date for completing an SE that we could discuss.

MEMBER REMPE: So you're saying that you do not want to share the March 2022 draft SE with ACRS at this time?

MR. HELVENSTON: I think we have to discuss that among the staff a little more.

MEMBER REMPE: Okay. I just only last week learned that, oh no, they've been submitting changes to the CP, and there have been some design changes. And I just think it might be good for us to kind of be aware of things a little bit earlier.

Anyway, it's up to Dave and you guys, I guess, to discuss that further.

CHAIRMAN PETTI: So, Ed, I would just really push for -- because at least you can get some chapters to us that's not -- from approved SER to letter is really tight. So, if we're seeing stuff early, even September 2022, I think that'll work because we've got multiple reviews going on in parallel.

MR. HELVENSTON: I understand. No, that's helpful feedback.

MEMBER MARCH-LEUBA: Yeah. What were you speaking to do between November 2022 and May '23? Just the OGC review?

MR. HELVENSTON: So, based on the schedule for this slide, yeah, that's correct. November is essentially when the staff would have the SE completed, and then the interim period until May 2023 is for the management and OGC reviews.

MEMBER MARCH-LEUBA: Yeah. We couldn't even go over the technical parts before it's too late change.

MR. HELVENSTON: Yeah, and we have to make sure we're in process, certainly, and in terms of -- because I understand there's things about the draft

SE, SE chapters going to ACRS being made public. So it's something we need to discuss with the staff and certainly with OGC as well, just to make sure that we're in process.

MEMBER REMPE: So it could be reference material. But it might help with reviewing some of the topical reports that are coming to us in the interim.

MEMBER MARCH-LEUBA: That too.

DR. BLEY: If you talk with our staff, you'll find there are arrangements we've made in the past to see documents beforehand. It's when we're coming to an ACRS meeting to discuss documents that they have to be made public.

MEMBER MARCH-LEUBA: The PSAR and your SER are completely non-proprietary, correct?

MR. HELVENSTON: The PSAR is non-proprietary; that's correct. The SER -- certainly, the staff will strive to issue a non-proprietary SER. I don't know if we've made a final decision on that yet, though.

MEMBER MARCH-LEUBA: At the minimum, you have to be sure non-proprietary mark-up.

MR. HELVENSTON: That's right. At a minimum, there will be a non-proprietary mark-up.

CHAIRMAN PETTI: Anything else, Ed? Is this the last slide, or --

MR. HELVENSTON: Yeah. This is the last slide for me.

CHAIRMAN PETTI: Okay.

MR. HELVENSTON: I can certainly take any other questions.

CHAIRMAN PETTI: No. You know what I'd like to do, though, Ed, is have a meeting with Weidong and you and whoever else, sort of at the project management level. And we've made some initial assignments of who is responsible for what, but now that I better understand based on what I heard today, I want to at least loop back with Weidong and maybe you guys to -- I see some natural groupings of chapters that could help accelerate the review and see if we can get on the same page there and make sure that I'm not missing something in terms of my thought process. So if we can do that, I think that would be good.

MR. HELVENSTON: Sounds good.

CHAIRMAN PETTI: So Weidong, if you'd try to set something up, thanks.

Okay. Members, any other comments before we go to the public?

MEMBER DIMITRIJEVIC: Sure. I would like to make a general comment, Dave.

CHAIRMAN PETTI: Sure. Sure.

MEMBER DIMITRIJEVIC: This is Vesna Dimitrijevic. After we have all of this discussion, in the end, I don't think that you guys should consider the big fact that your review is based on this consecutive significance, because we don't know too much. We don't know anything about risk, and we know very little about safety significance.

So I would propose that you phrase this a little more in the sense that as you have minimum amount of regulation based on 104(c) and some estimation of the safety impacts -- because in this moment, as we see in one of your last slides, that the most of the safety issue -- this is all qualitative, and most of -- even if you look in mitigation systems -- as we can see, light-water reactor, very often, non-safety systems show to be safety significant. So being safety or non-safety doesn't necessarily preclude that.

So I will rephrase this as, based on safety right significance. It's not going to be based really on the realistic safety significance. So that's my general comment.

MR. HELVENSTON: Well, thank you for that.

CHAIRMAN PETTI: Other comments, members?

Okay. Then let's go to public comment. If you're a member of the public, if you're on the phone, star-6. If you're on Teams, raise your hand, and we'll recognize you and make your comment.

Okay. Not hearing any, I think that means we are done. I want to thank both the staff and Kairos. It was a most enlightening afternoon, lots of ground to cover in a short amount of time. This went about as I expected it would. We always have lots of perspectives and interest in different areas, and you can see we just love to probe because that's what we like to do. But we really look forward to seeing this come to fruition given its significance as the first advanced non-light-water reactor.

With that, I will say we are done and end our meeting. Everybody have a good weekend, and members, we'll see you at May Full Committee. Thank you all.

(Whereupon, the above-entitled matter went off the record at 5:26 p.m.)

KP-NRC-2204-007

April 15, 2022

Docket No. 50-7513

US Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Kairos Power LLC
Presentation Materials for Kairos Power Briefing to the Advisory Committee on Reactor Safeguards, Kairos Power Subcommittee, on Design Overview for the Hermes Non-Power Reactor

This letter transmits presentation slides for the April 21, 2022, briefing to the Advisory Committee for Reactor Safeguards (ACRS), Kairos Power Subcommittee. At the meeting, Kairos Power will provide an overview of the design of the Hermes non-power test reactor which is currently under NRC staff review for a construction permit. This briefing is intended to provide a high level overview of the Hermes design prior to the ACRS review of the Hermes PSAR.

The content of this information is non-proprietary; Kairos Power authorizes the Nuclear Regulatory Commission to reproduce and distribute the submitted content, as necessary, to support the conduct of their regulatory responsibilities.

If you have any questions or need additional information, please contact Drew Peebles at peebles@kairospower.com or (704) 275-5388, or Darrell Gardner at gardner@kairospower.com or (704) 769-1226.

Sincerely,

Darrell Gardner, for

Peter Hastings, PE
Vice President, Regulatory Affairs and Quality

Enclosures:

1) Presentation Slides for the April 21, 2022, ACRS Kairos Power Subcommittee Briefing

xc (w/enclosure):

William Kennedy, Acting Chief, NRR Advanced Reactor Licensing Branch
Benjamin Beasley, Project Manager, NRR Advanced Reactor Licensing Branch
Weidong Wang, Senior Staff Engineer, Advisory Committee for Reactor Safeguards

Enclosure 1

Presentation Slides for the April 21, 2022
ACRS Kairos Power Subcommittee Briefing

Hermes Design Overview

PRESENTATION FOR THE ADVISORY COMMITTEE
ON REACTOR SAFEGUARDS, KAIROS POWER SUBCOMMITTEE

APRIL 21, 2022

Kairos Power's mission is to enable the world's transition to clean energy, with the ultimate goal of dramatically improving people's quality of life while protecting the environment.

In order to achieve this mission, we must prioritize our efforts to focus on a clean energy technology that is *affordable* and *safe*.

Agenda

- Introduction
- Fuel/Core Design
- Reactor Vessel and Internals
- Heat Transport & Pebble Handling and Storage
- Structures
- I&C and Electrical
- Safety Case

Introduction

DREW PEEBLES – LICENSING MANAGER, SAFETY

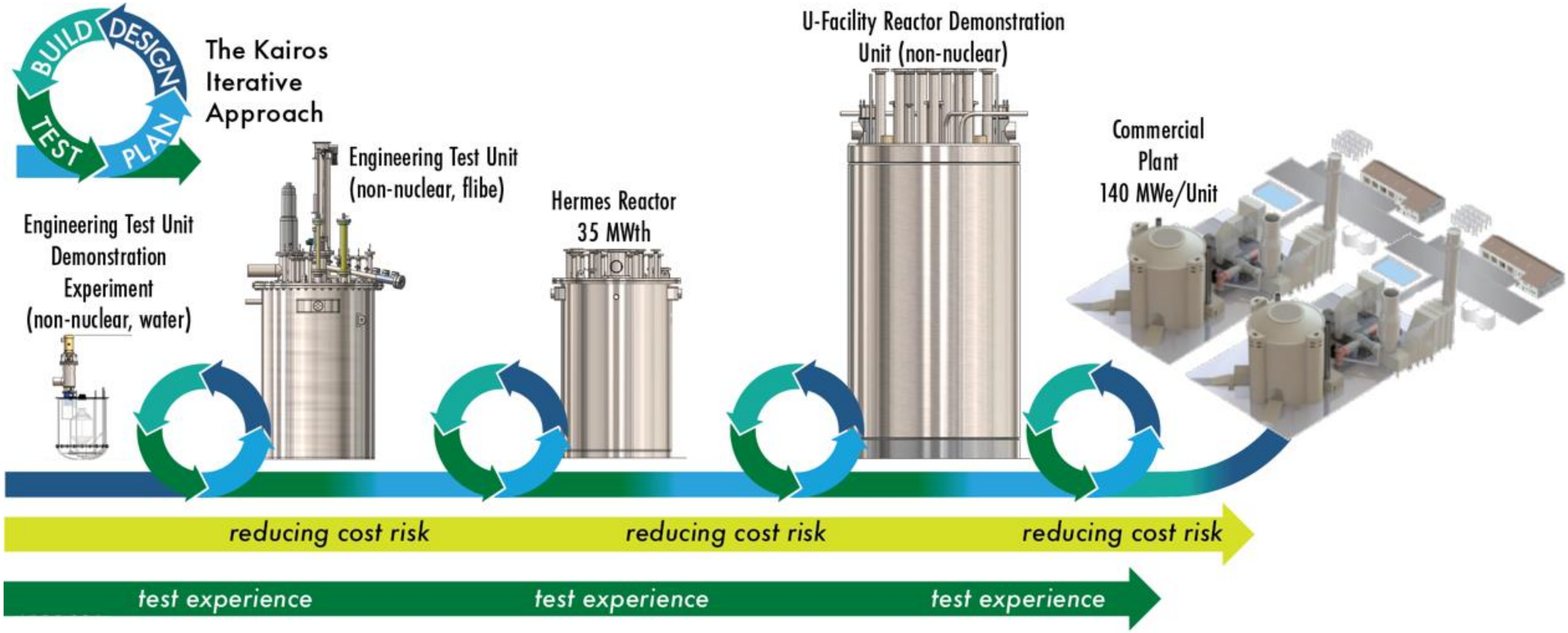
Introducing Kairos Power

- Nuclear energy engineering, design and manufacturing company ***singularly focused*** on the commercialization of the fluoride salt-cooled high-temperature reactor (FHR).
  - Founded in 2016
  - Current Staffing:
    - 269 Employees (***and growing***)
    - ~90% Engineering Staff
- Private funding commitment to engineering design and licensing program and physical demonstration through nuclear and non-nuclear technology development program.
- Schedule driven by the goal for U.S. commercial demonstration by 2030 (or earlier) to enable rapid deployment in 2030s.
- Cost targets set to be competitive with natural gas in the U.S. electricity market.

Kairos Power Headquarters

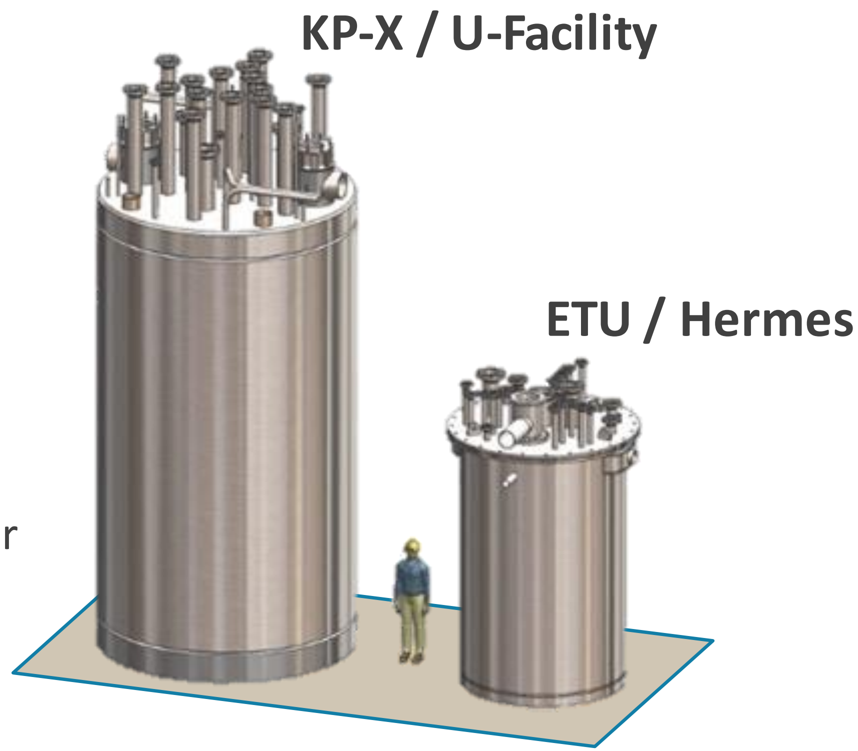
Kairos Power Team

Kairos Power Design Approach



Kairos Power Hermes Reactor Overview

- What?
  - A low power demonstration reactor that will prove Kairos Power's capability to deliver low-cost nuclear heat
- Why?
  - **Cost:** Establish competitive cost through iterative learning cycles
  - **Supply Chain:** Advance the supply chain for KP-FHR specialized components and materials while vertical integrating critical systems
  - **Design / Test:** Deliberate and incremental risk reduction
  - **Licensing Approach:** NRC will license Hermes as a non-power reactor and facilitate licensing certainty for KP-FHR
  - **Operations:** Provide a complete demonstration of nuclear functions including reactor physics, fuel and structural materials irradiation, and radiological controls



Hermes will ultimately demonstrate the U.S. aptitude to license an advanced reactor in a timely manner

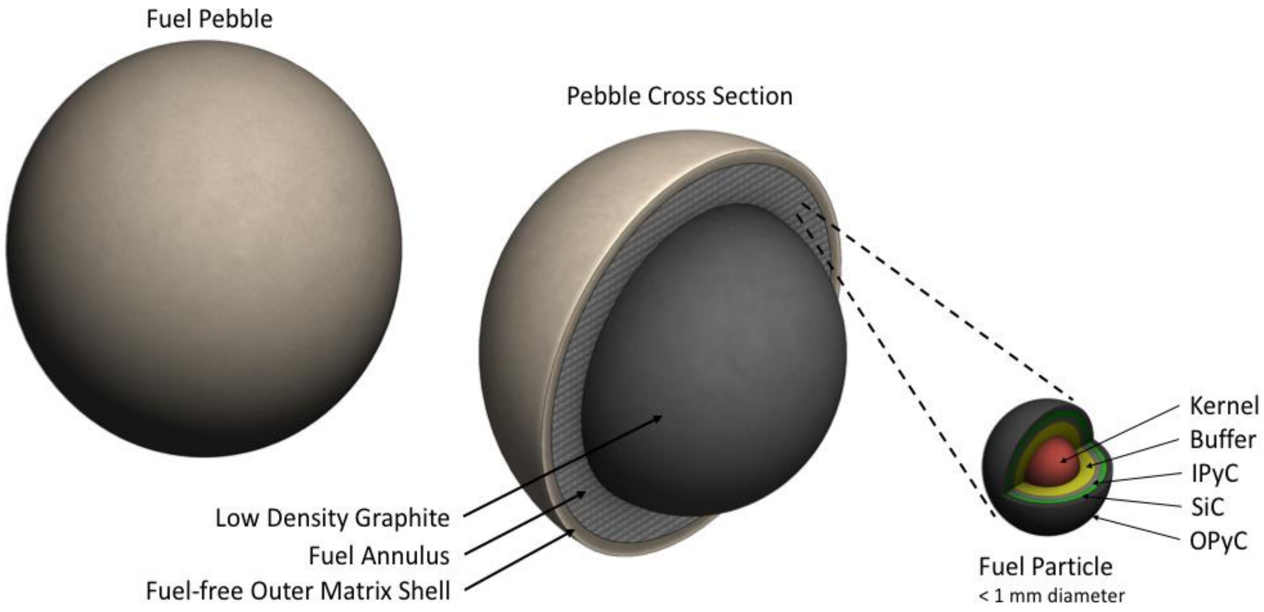
Fuel/Core Design

BRANDON HAUGH – SR. DIRECTOR, MODELING & SIMULATION

NADER SATVAT - MANAGER, REACTOR CORE DESIGN

KP-FHR Uses TRISO Fuel in Pebble Form

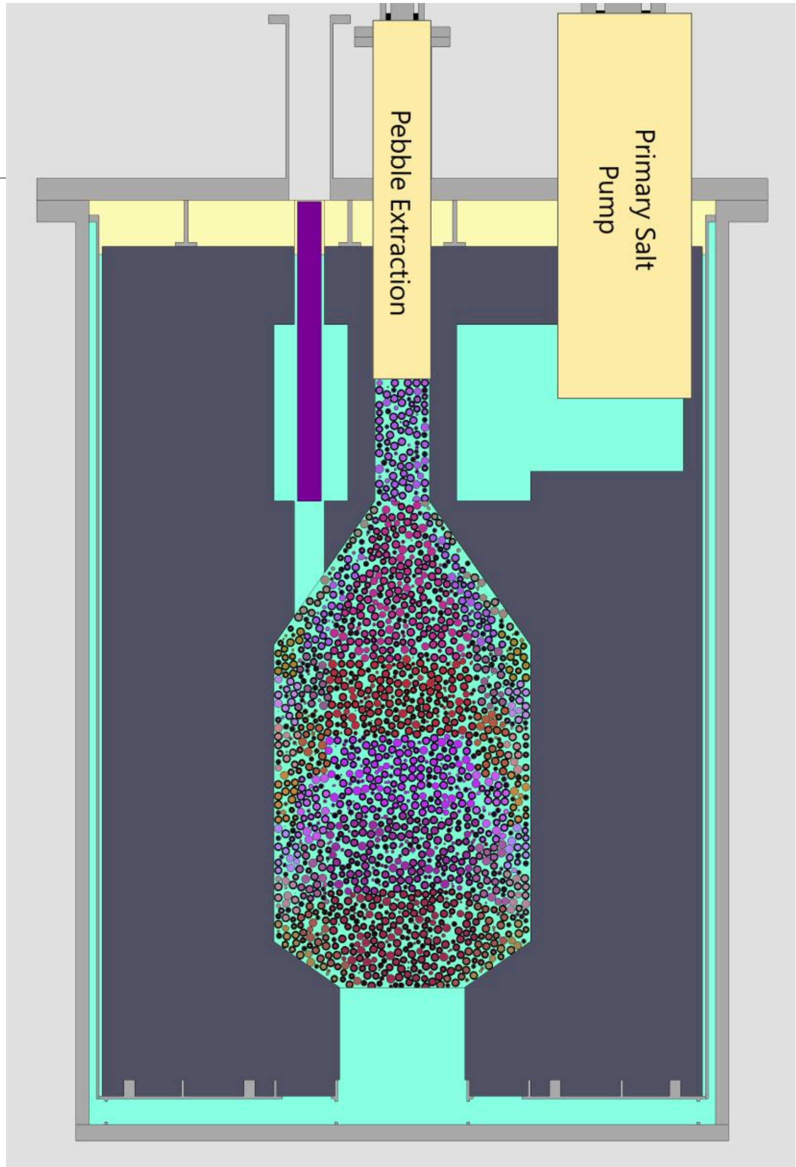
- Fuel Pebble (3 Regions):
  - Innermost portion is a low-density carbon matrix core
  - Fuel annulus - Tri-structural isotropic (TRISO)-coated fuel particles embedded in a carbon matrix
  - Fuel-free carbon matrix shell
- Fuel qualification leverages U.S. DOE Advanced Gas Reactor program
- Core design is a pebble bed concept within a graphite reflector
  - Pebbles are positively buoyant in Flibe
  - Mixture of fuel and moderator pebbles operates with optimal moderation



4.0-cm diameter, annular fuel pebble is the same size as a ping-pong ball

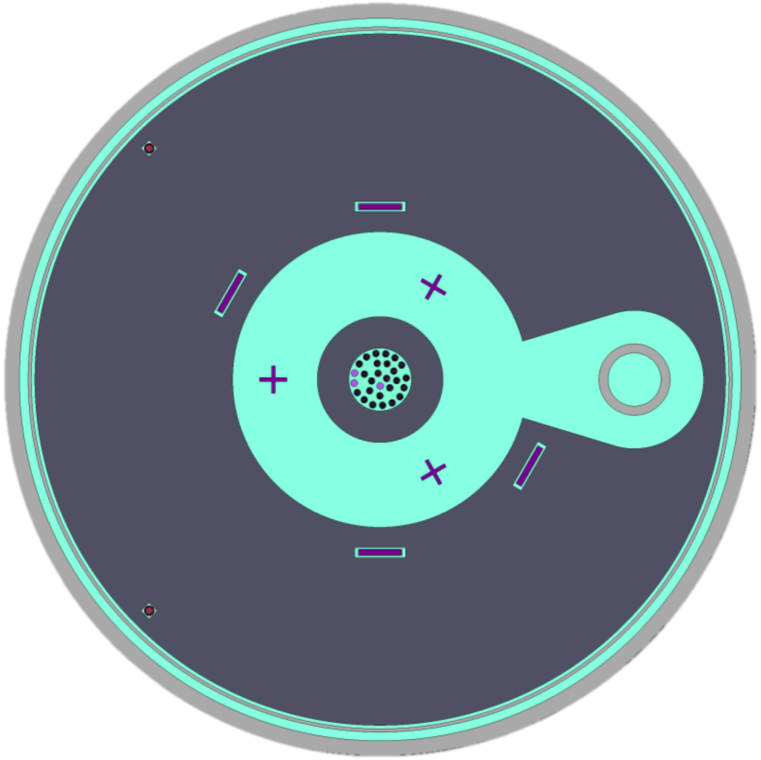
Hermes Core Design

Power:	• 35 MW_{th}
Fuel Cycle:	• 190 days average residence time • 4-6 passes • Discharge burnup 6-8% FIMA
Safety Parameters:	• Overall negative temperature reactivity coefficients • Negative fuel and moderator temperature reactivity coefficients • Negative coolant temperature, and void coefficients
Method for Calculation:	• High-fidelity Serpent 2 and KPACS (Serpent 2/Shuffling)
Power Profile:	• Average Power per pebble = ~1000 W/pebble • Pebble Peaking factor ~2
Coolant:	• Li-7 enrichment level and carbon to heavy metal atom ratio aligned to provide desired temperature reactivity coefficient

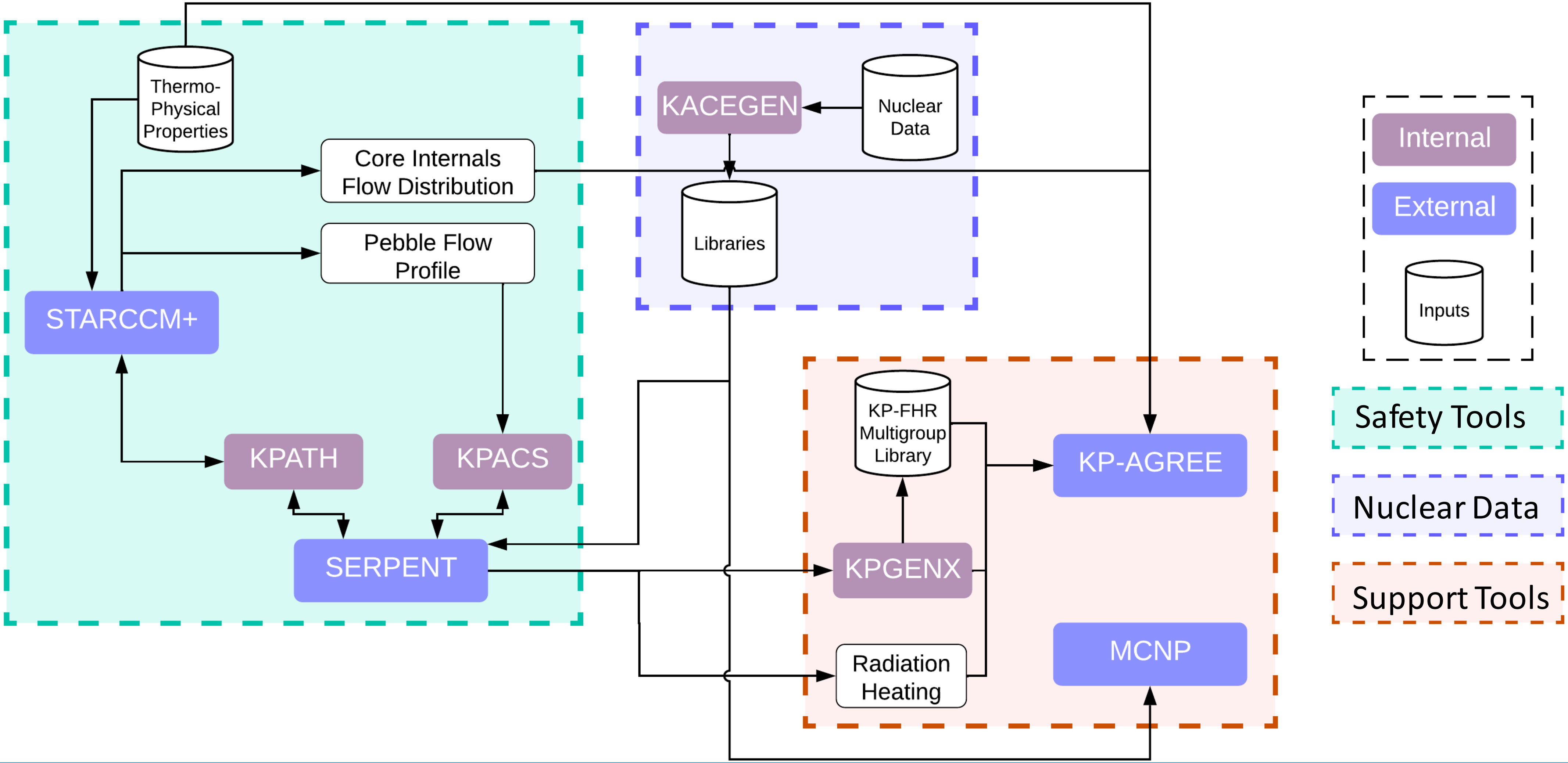


Reactor Control

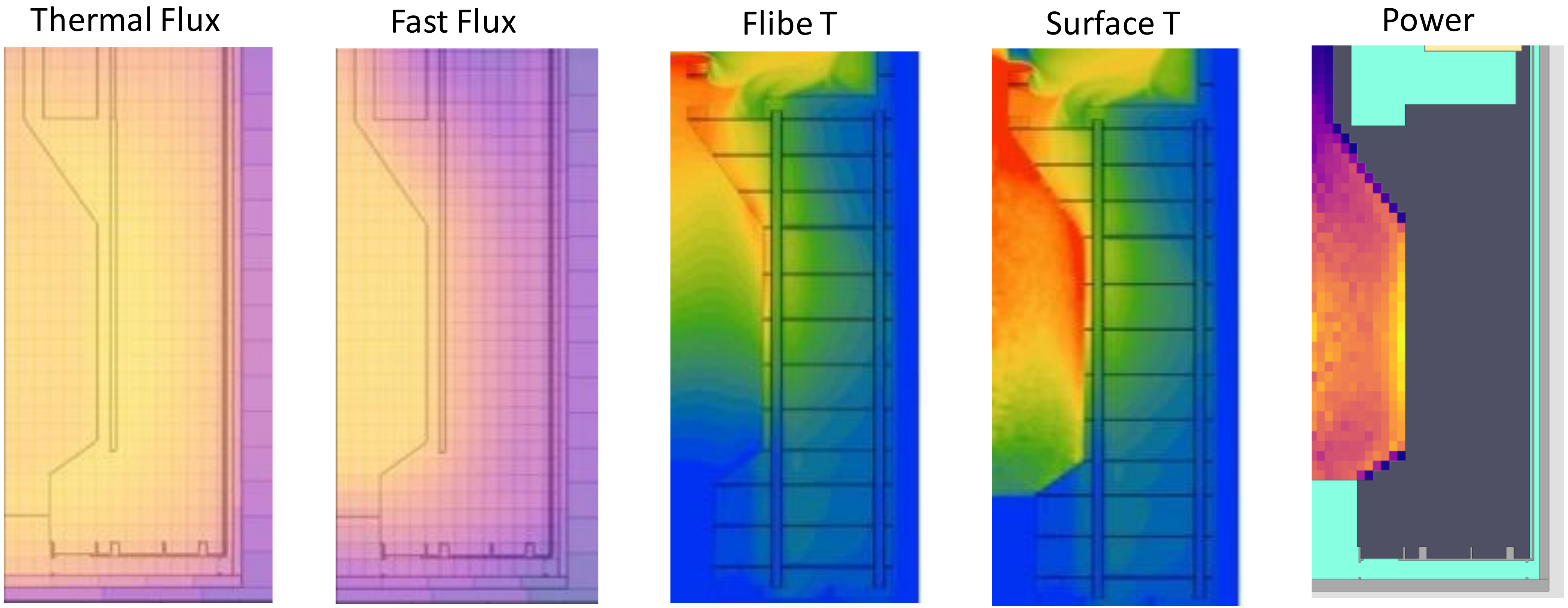
Diversity:	• Reactivity Control System (RCS) • Reactivity Shutdown System (RSS)
Shutdown Margin (SDM) Analysis:	• Compensate power defect, full xenon decay, operational excess reactivity, and B_4C depletion • Single, most reactive rod failure • SDM to k_{eff} of 0.99
Sources of Operational Excess Reactivity:	• Core composition • Compensate change power levels or manage other transients
Method for Calculation:	• High-fidelity coupling tool, KPATH (Serpent 2/Star-CCM+)
Other notes:	• Drive mechanism sets limit on withdrawal rate (rate of reactivity insertion) • KP-FHR has a strong (and prompt) Doppler feedback to reduce regular use of the RCS



Core Design Methodology



Representative Information



1×10^{8} – 5×10^{14} $n/cm^2 \cdot s$

550-700 °C

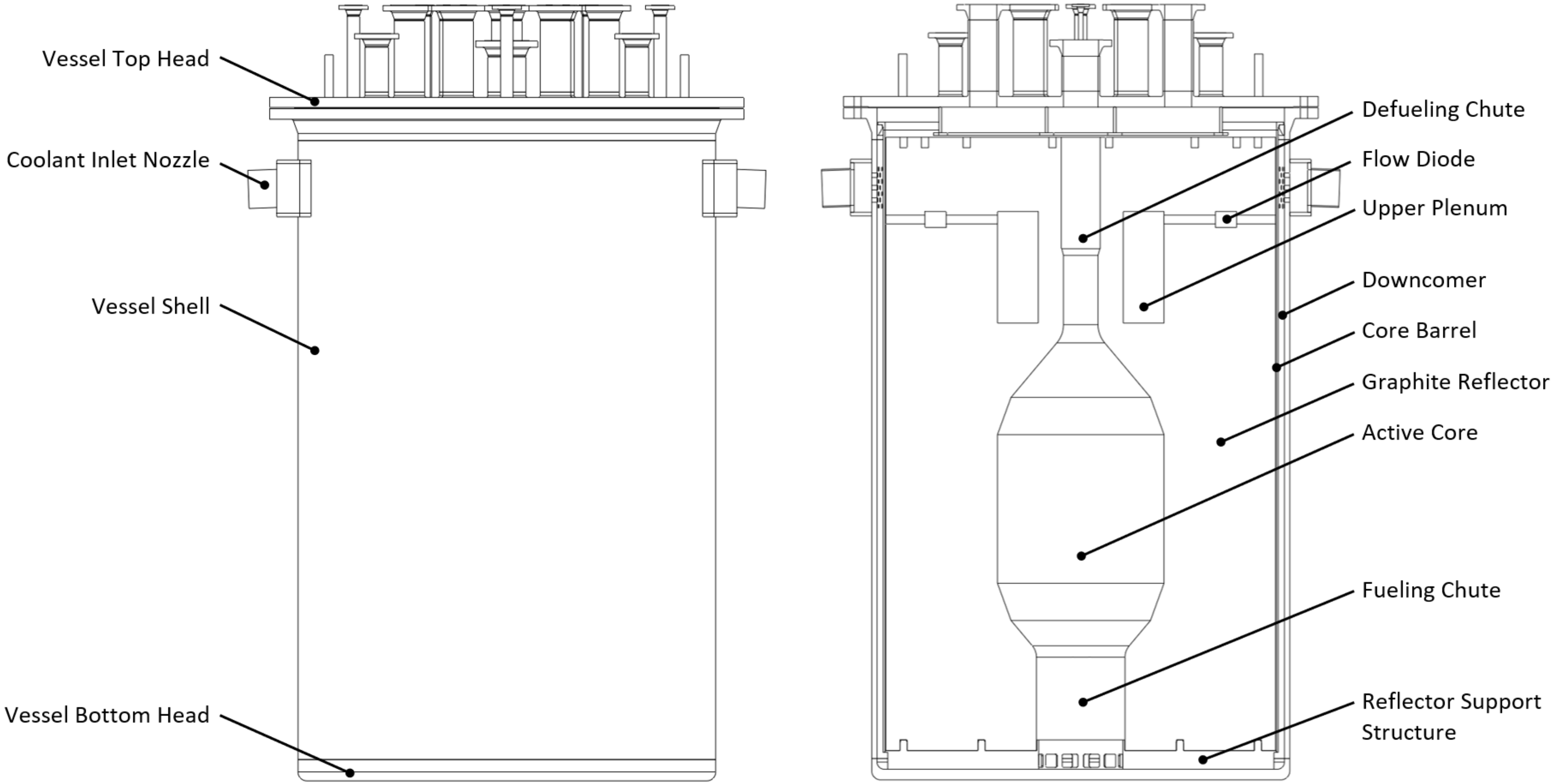
550-700 °C

0 – 26 W/cm^3

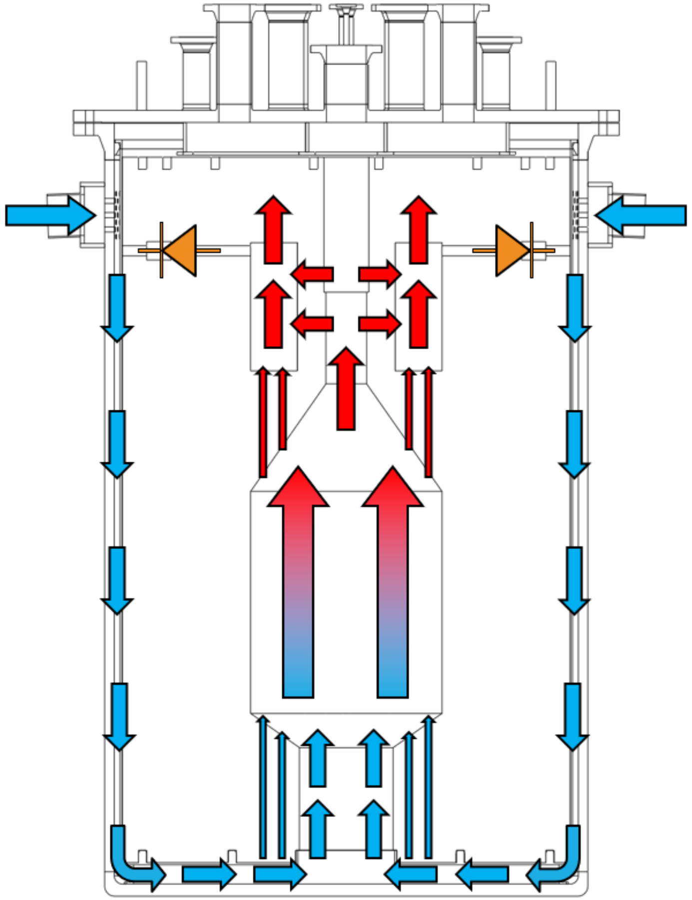
Reactor Vessel and Internals

ODED DORON – SR. DIRECTOR, REACTOR SYSTEM DESIGN

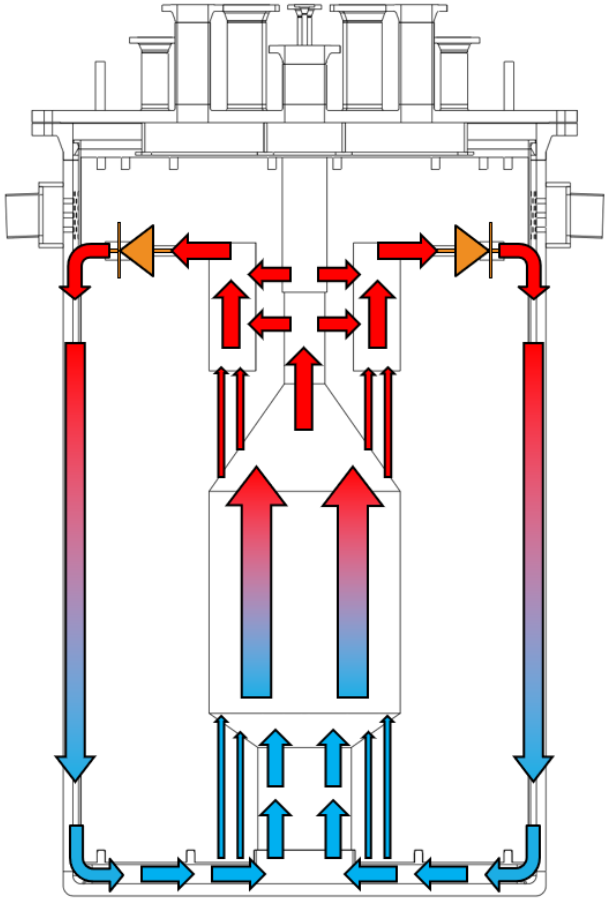
Reactor Vessel and Internals Overview



Hermes Coolant Circulation Path Overview

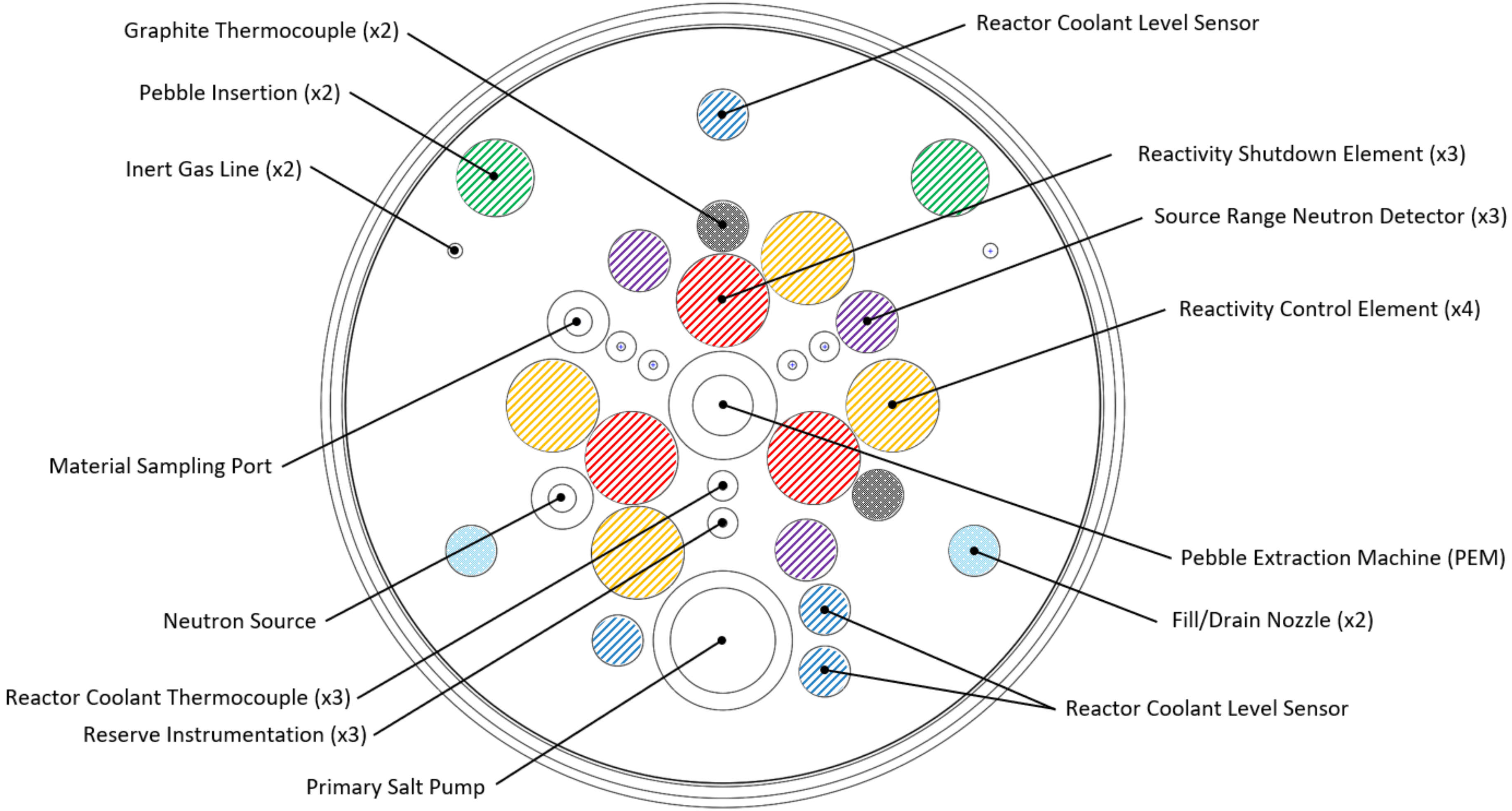


(a) Normal Operation Coolant Flow Path

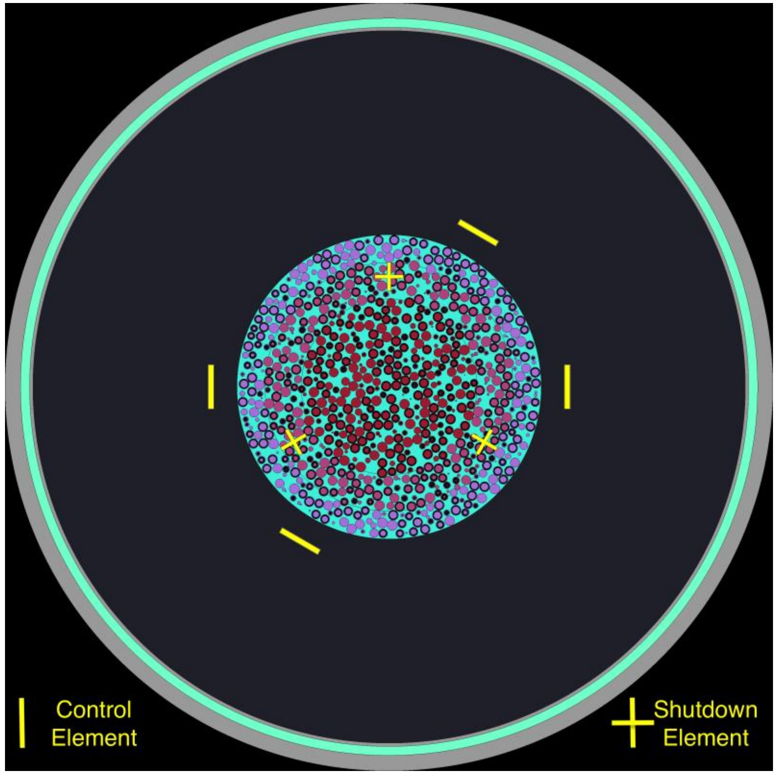


(b) Natural Circulation Coolant Flow Path

Hermes Head Layout



Hermes Reactivity Control and Shutdown System



Hermes Core Layout

3x inbed shutdown elements

4x excore control elements

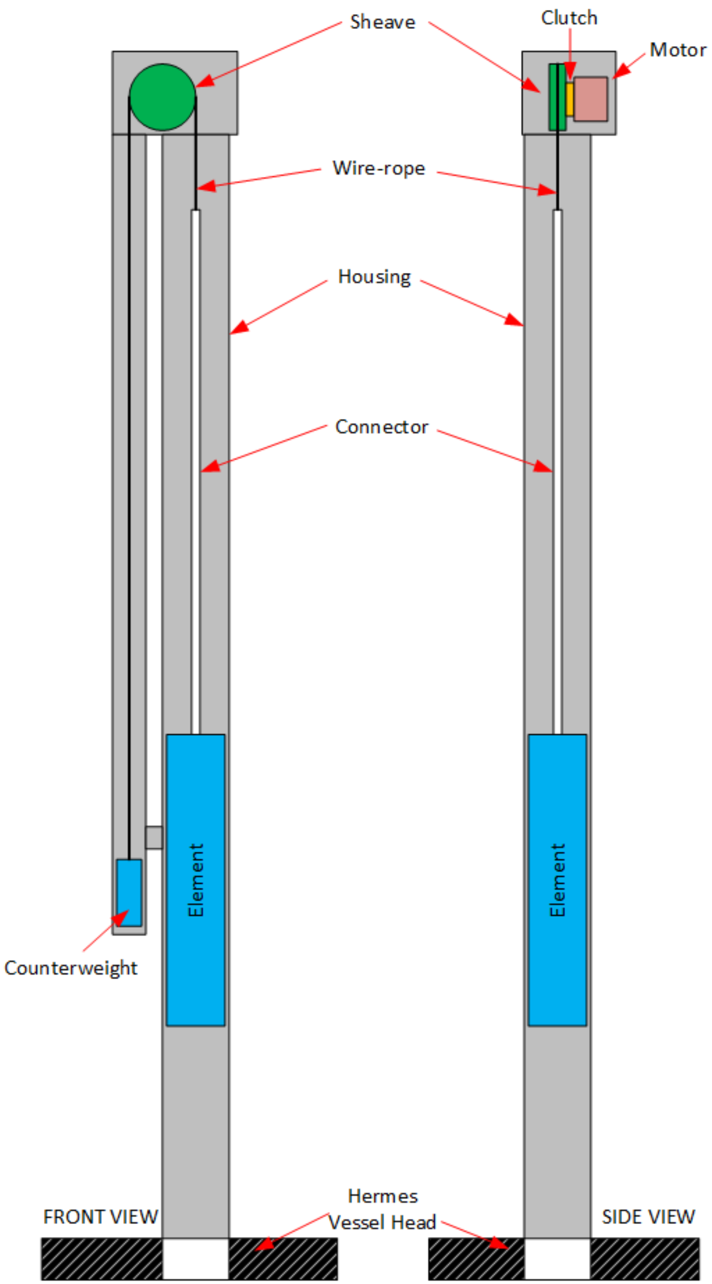
Drive Mechanism

Motor driven sheave to position element

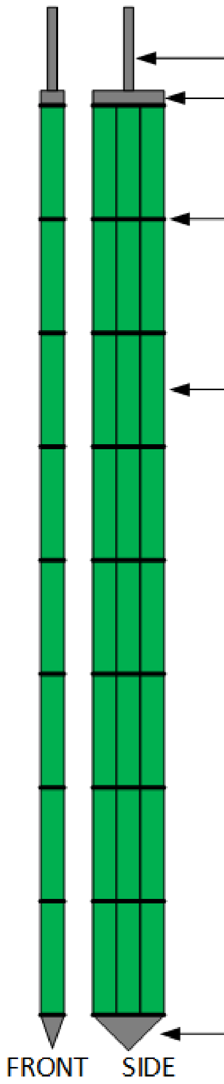
Release Mechanism

#1: Electromagnetic clutch

#2: Motor isolation

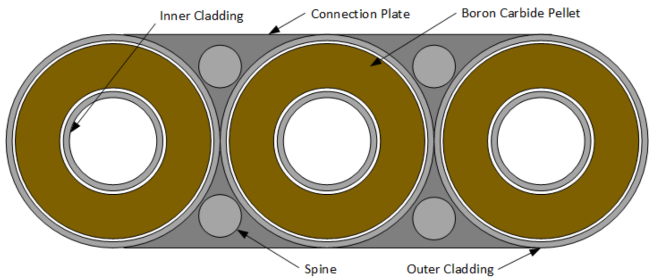


Control Element

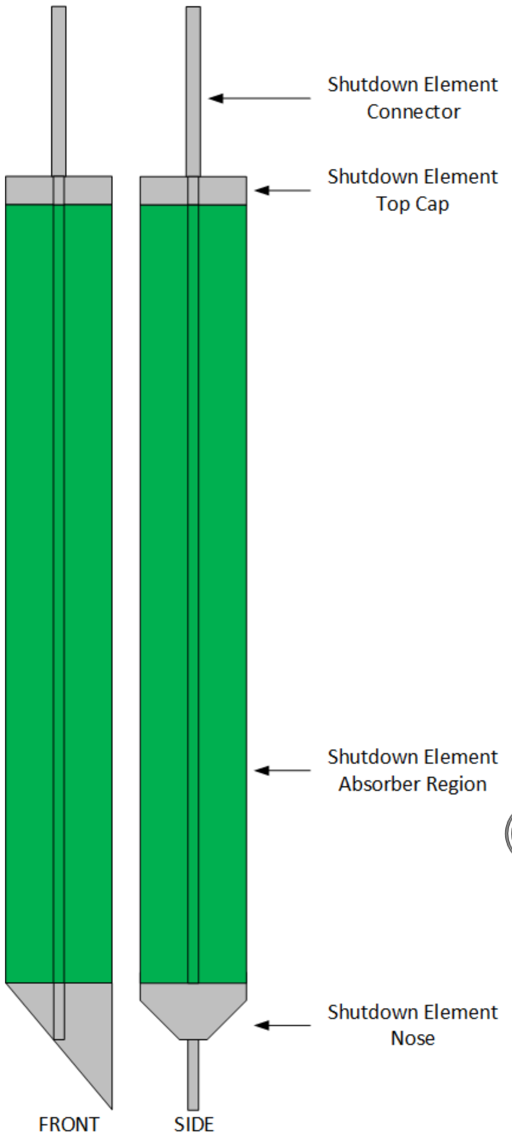


<u>Control Element</u>

Segmented Annular Design
Individual Capsules
Argon fill
Absorber: B4C
Cladding: SS-316H

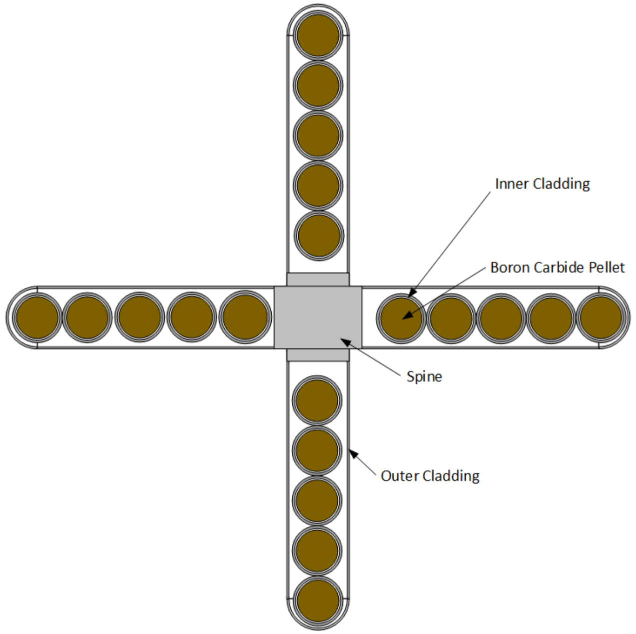


Shutdown Element



<u>Shutdown Element</u>

Cruciform Design
Inner Cladding contains absorber
Argon fill
Absorber: B4C
Cladding: SS-316H



Heat Transport & Pebble Handling and Storage

NICOLAS ZWEIBAUM – DIRECTOR, SALT SYSTEMS DESIGN

Primary Heat Transport System (PHTS) – Overview

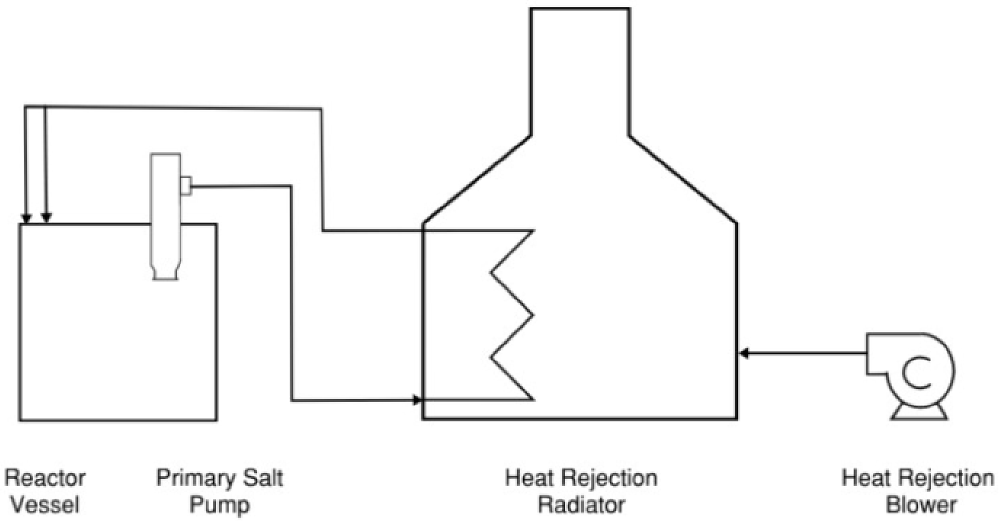
- The Primary Heat Transport System (PHTS) is responsible for transporting heat from the reactor to the ultimate heat sink (environmental air) during power operation and during normal shutdown
- The PHTS operates near atmospheric pressure and does not provide a safety-related heat removal function (see Decay Heat Removal System)
- The safety-related hot leg anti-siphon feature is performed by the Primary Salt Pump downward-facing inlet (the pump being supported in position by the Reactor Vessel upper head)
- Additionally, the PHTS provides for the following functions:
  - Contain and direct the reactor coolant flow between the reactor vessel and the heat rejection subsystem
  - Manage thermal transients (overall thermal balance) occurring as part of normal operations
  - Ensure minimum acceptable temperatures in the PHTS through make-up heating as necessary
  - Provide capability to drain the PHTS to reduce parasitic heat loss during over-cooling transients
  - Provide for in-service inspection, maintenance, and replacement activities

PHTS – System Makeup

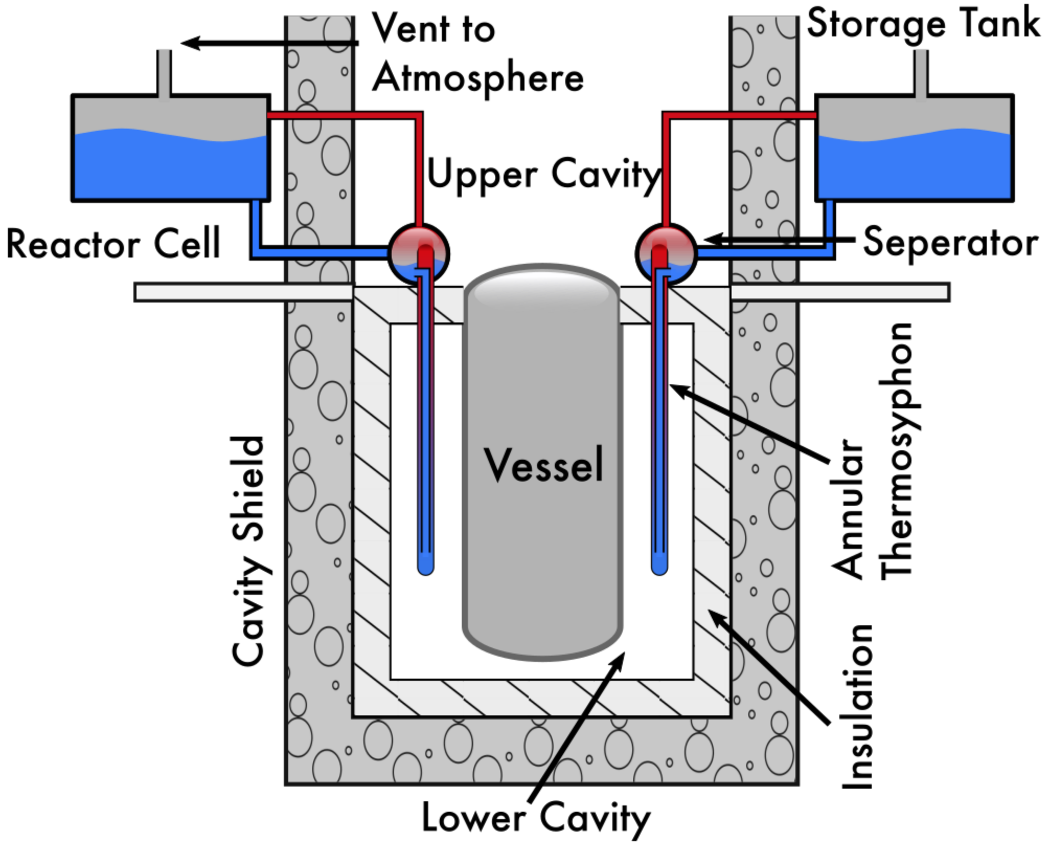
- Reactor Coolant
  - Flibe
- Primary Salt Pump (PSP)
  - Variable speed, cartridge style pump located on the reactor vessel head; inlet extends downwards through the Reactor Coolant free surface
- Heat Rejection Subsystem (HRS)
  - Provides for heat transfer from the reactor coolant to the atmosphere
  - Consists of the heat rejection radiator, heat rejection blower, and associated ducting and thermal management
- Primary Loop Piping
- Primary Loop Thermal Management
  - Provides non-nuclear heating and insulation to the PHTS as needed for various operations

PHTS – High Level Description

Parameter	Value
Thermal duty	35 MWth
Number of HRRs	1
Number of hot legs	1
Number of cold legs	2
Primary loop line size	8-12 in nominal pipe size
HRR inlet coolant temperature	600-650°C
HRR outlet coolant temperature	550°C
Nominal flow rate	210 kg/s
PHTS design pressure	525 kPa(g)



Decay Heat Removal System (DHRS) – Overview



Purpose: Vessel protection during postulated events for which the primary heat transport system (PHTS) is unavailable

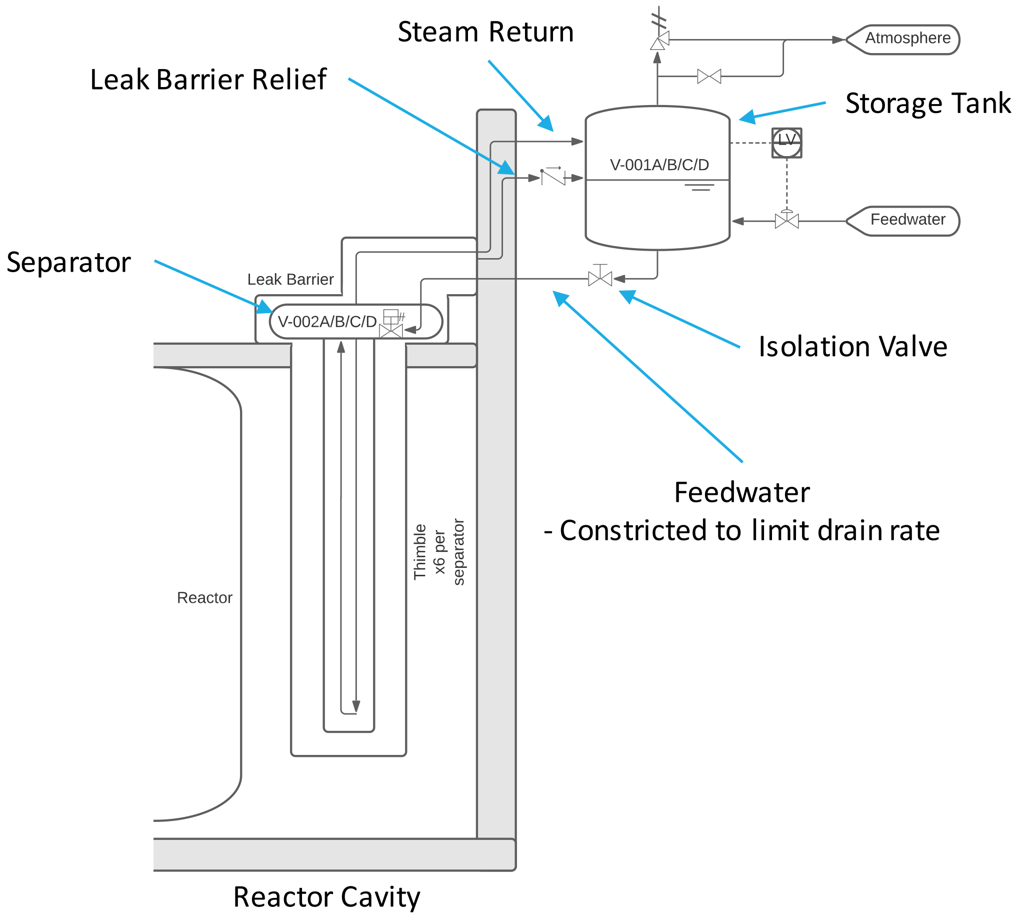
Operation: In-vessel natural circulation coupled to a passive water-based, ex-vessel system via thermal radiation and convection

- Continuous direct boil-off when estimated decay loads exceed parasitic losses
- Shutoff and isolated for low power levels (heat removal via parasitic losses only)
- No change of state on reactor event initiation

Load: Removal rate is a function of vessel temperature

- Due to physics of thermal radiation heat transfer

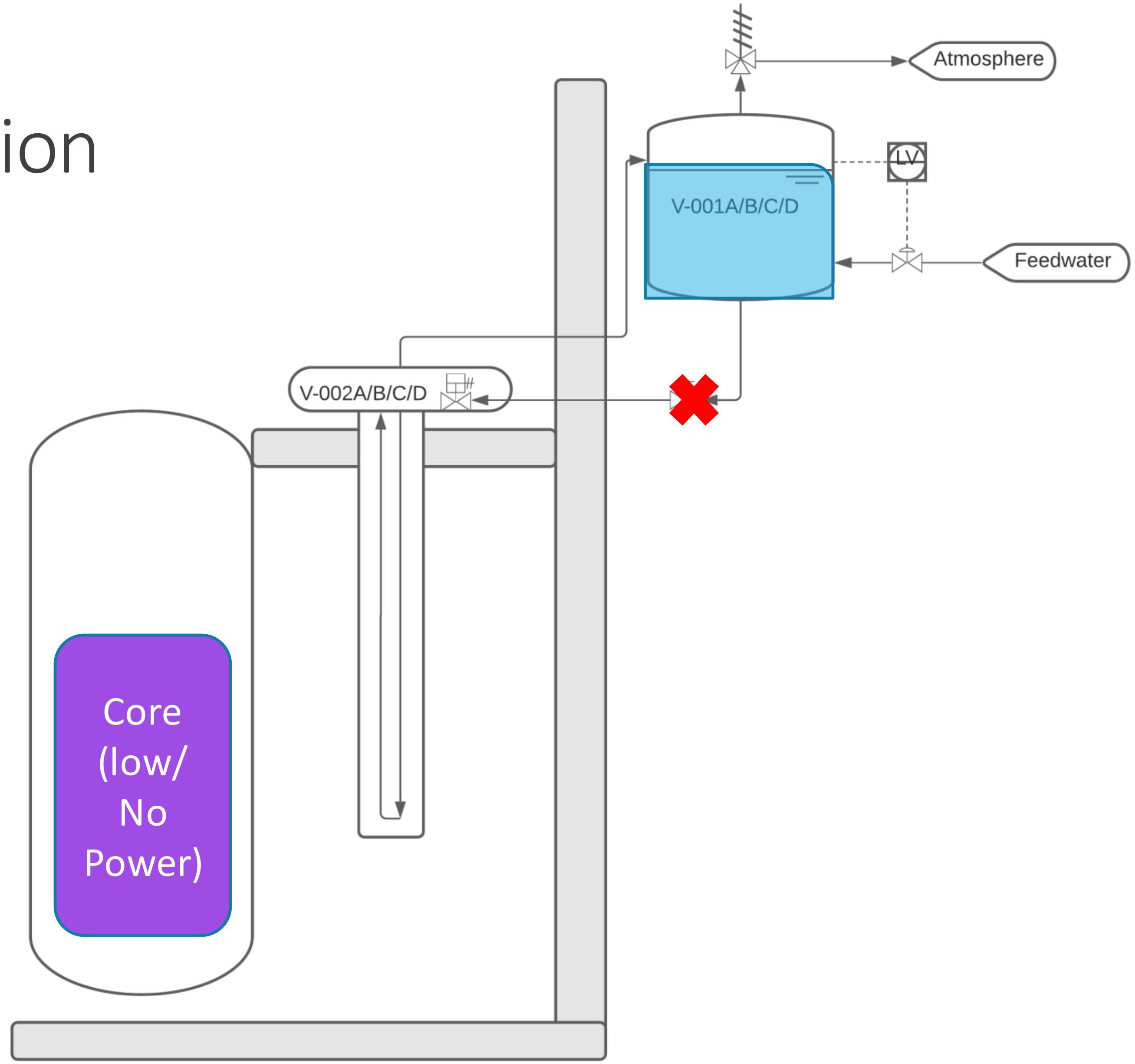
DHRS – Process Flow Diagram



- DHRS does not directly interact with the primary coolant
- No change of state on onset of postulated events
  - Always-on operation for set power levels
- Parallel and independent cooling pathways
  - Four independent cooling loops
  - Only three loops required to meet cooling demand
- Dual-walled for leak prevention and detection
  - Continued heat removal in the presence of a leak
- Active component (isolation valve) failures do not introduce failures in heat removal
  - Isolation valve fails in place (an operating system continues to operate)
  - Float valve nominally fails open

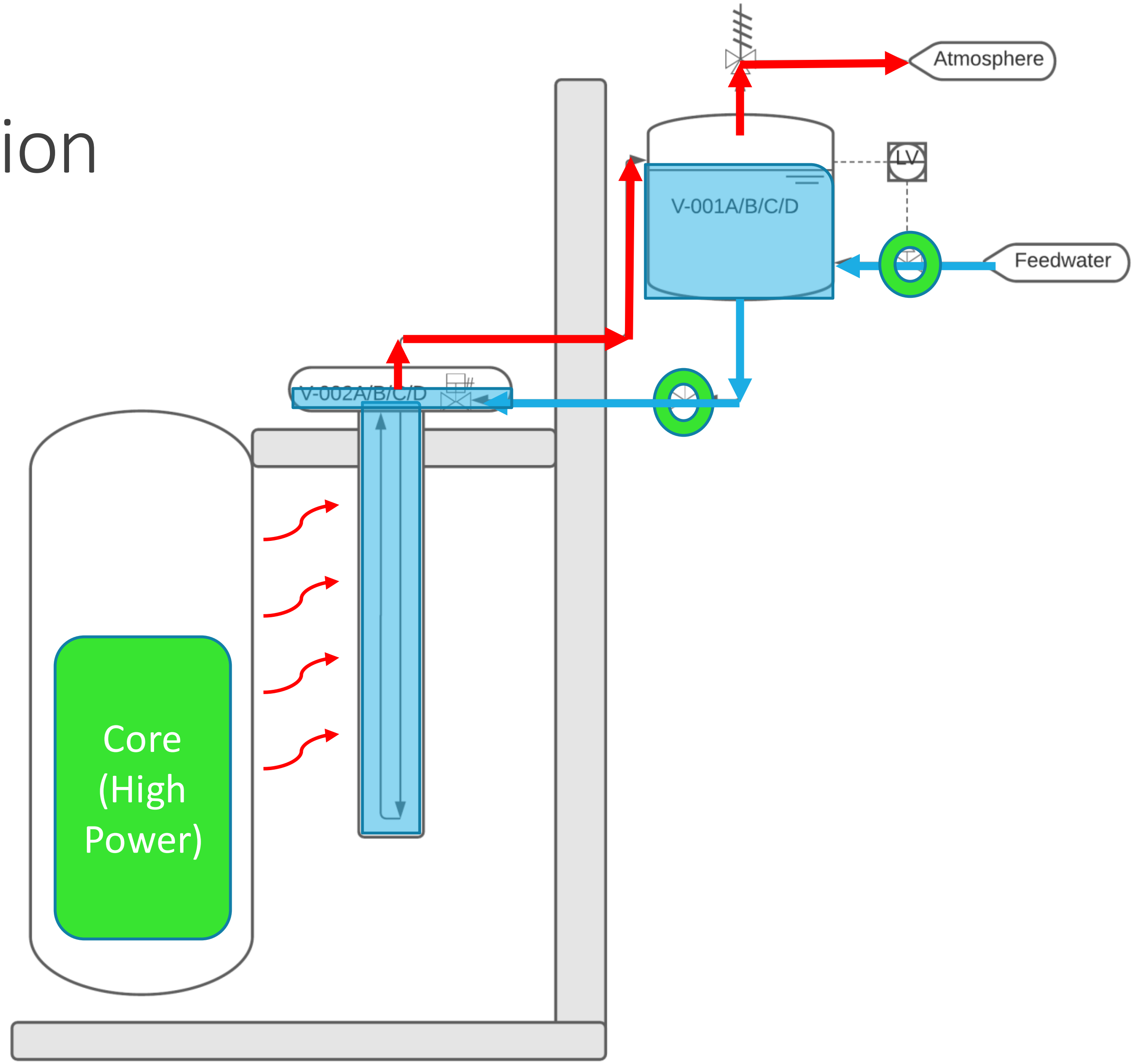
DHRS – Operation

Normal Operation
(DHRS deactivated)



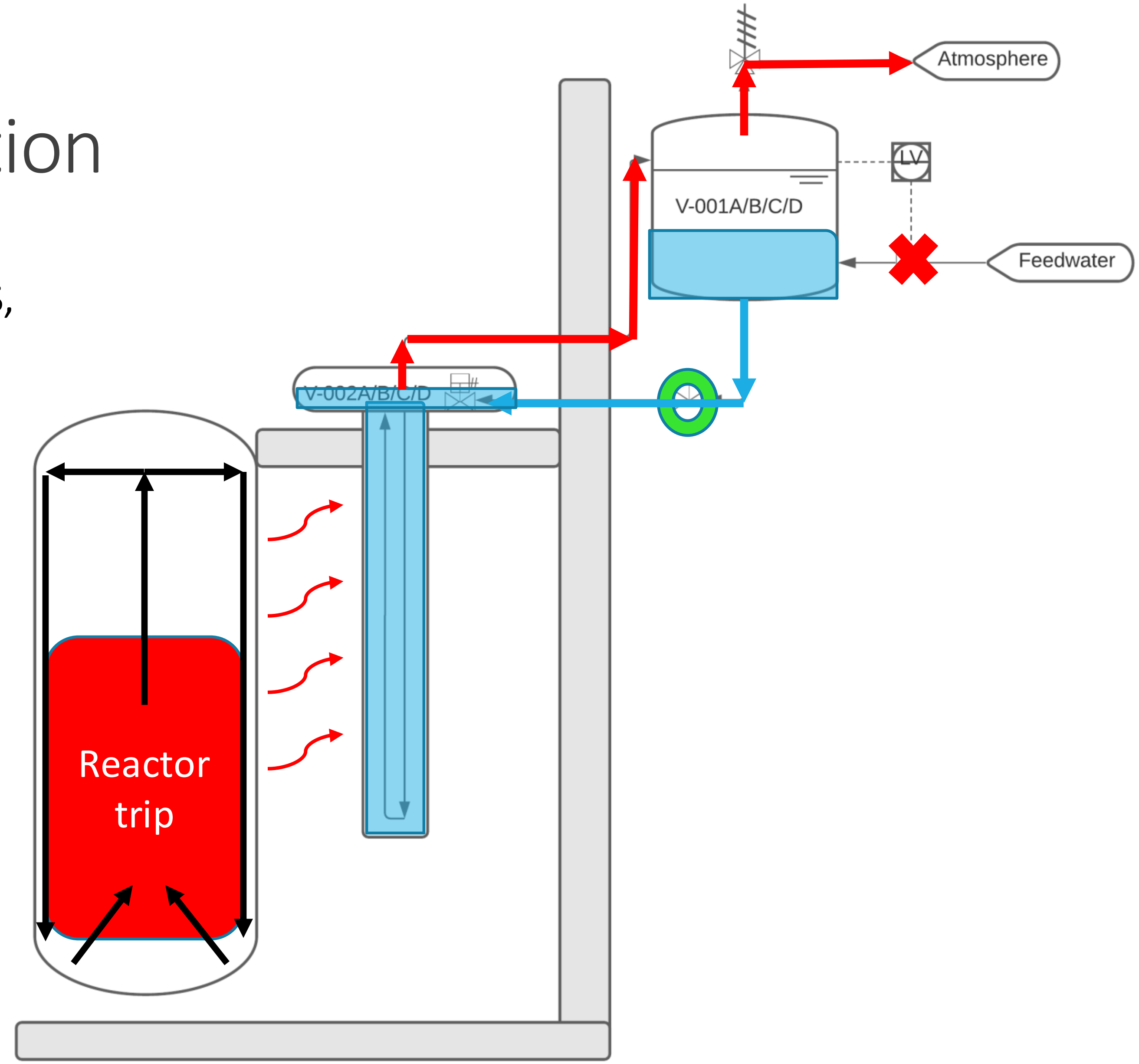
DHRS – Operation

Normal Operation



DHRS – Operation

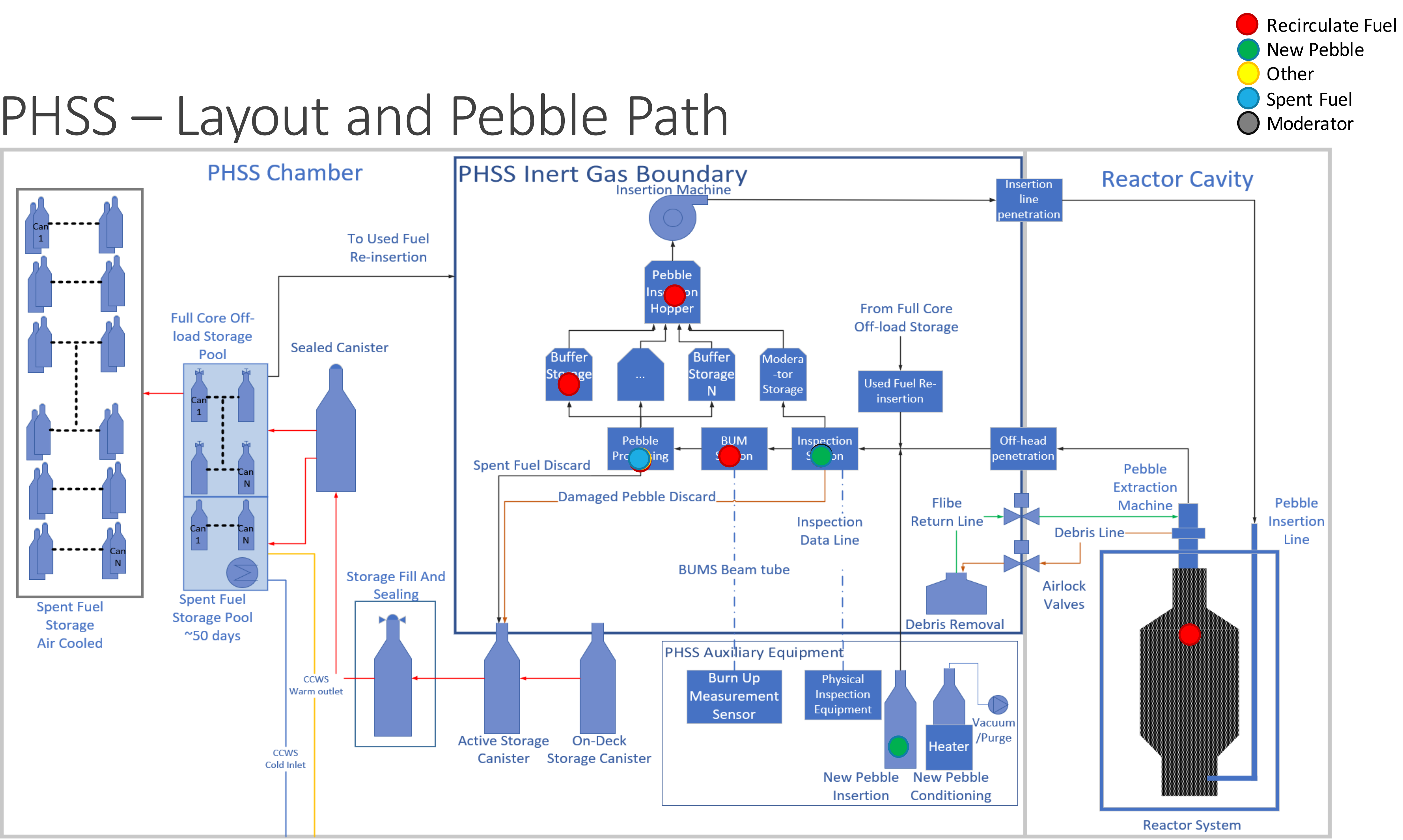
Transient Event
(unrecoverable loss of PHTS,
loss of electrical power,
loss of feedwater)



Pebble Handling and Storage System (PHSS) – Overview

- Responsible for handling of fuel in Hermes, from initial on-site receipt, in-process circulation, and final on-site storage
- Major components of the system:
  - Pebble Extraction Machine (PEM): single screw for removing pebbles from molten salt
  - Pebble Inspection System: performs flaw detection and burn-up measurement of removed pebbles
  - Processing System: sorts pebbles into appropriate buffer storage channel based on pebble type
  - Insertion System: stepper wheel feeder mechanism that inserts pebbles into the reactor via an in-vessel insertion line
  - Storage System Canister: stores ~2,000 damaged or spent fuel pebbles in a non-critical configuration
  - Storage Cooling Area: passively cooled, in-building storage area for spent fuel canisters
  - New Pebble System: stores fresh fuel and prepares fuel for circulation via a high-temperature bakeout

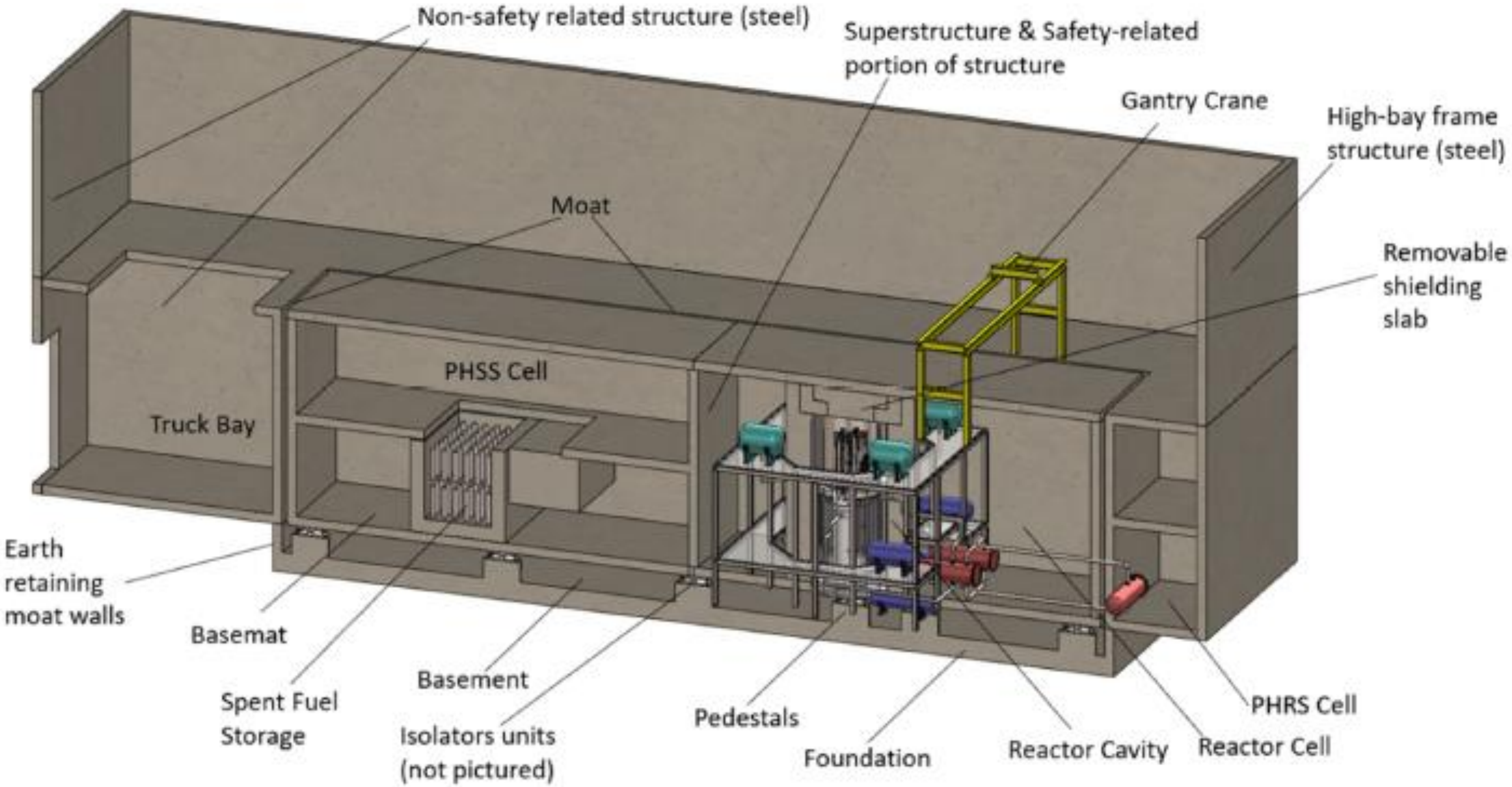
PHSS – Layout and Pebble Path



Structures

BRIAN SONG – MANAGER, CIVIL STRUCTURES

Reactor Building Layout



Meteorological Loads

- Design considers rain, snow, wind, tornado and wind-borne missiles for site.
- Safety-related reactor building designed without crediting non-safety-related exterior shell for protection from snow, wind, rain, and missile loads.
- Exterior “shell” of safety-related reactor building designed with concrete thickness to protect safety-related structures, systems, and components (SSCs) from high-wind missiles, including debris from potential damage of non-safety-related reactor building.

Flood loads

- Safety-related SSCs will be protected from internal flood (spray and accumulation) with shields, curbs, drains, etc.
- Safety-related Reactor Building protects safety-related SSCs from credible external flood.

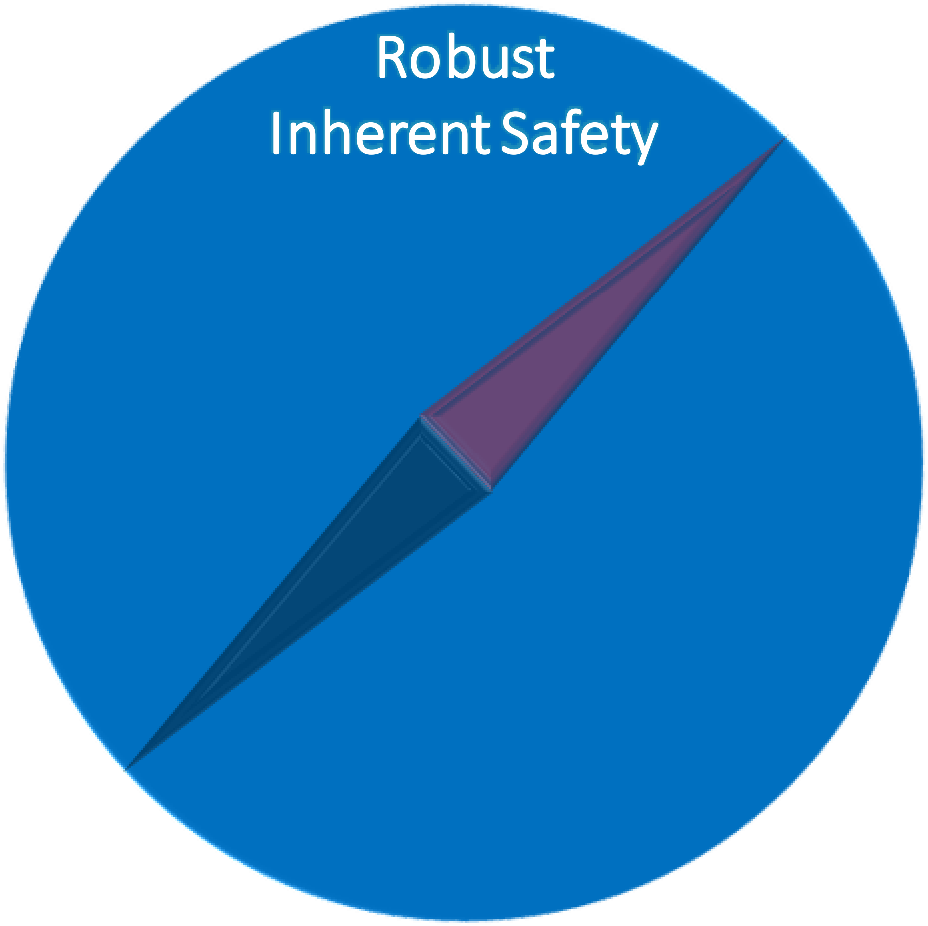
Seismic Loads

- Using risk-informed performance-based insights to define seismic design criteria (i.e. ASCE 43-19, SDC 3)
  - Seismic design basis earthquake based on site-specific seismic hazard considering other recent and nearby seismic hazard analyses and site-specific geotechnical characteristics.
- Safety-related Reactor Building incorporates spring/dashpot seismic isolation system, which lowers seismic demands on safety-related reactor building and safety-related SSCs in both horizontal and vertical directions.
- Moat and flex connections accommodate displacements of isolated safety-related reactor building.
- Safety-related portion of the Reactor Building will be represented by a three-dimensional finite-element model developed in accordance with Chapter 3 of ASCE 4-16.

Instrumentation & Controls and Electrical Systems

ANTHONIE CILLIERS – DIRECTOR, INSTRUMENTATION, CONTROLS AND ELECTRICAL

Instrumentation & Controls and Electrical System Design Relies on the Following Systems



Plant protection and control

- Reactor Protection System (RPS)

Safety Hazard Intervention and Event Limiting Defense

- Plant Control System (PCS)

System with Operational Reliability and Diagnostics

- Intelligent Health Monitoring

Health Evaluation and Analysis in Real-Time

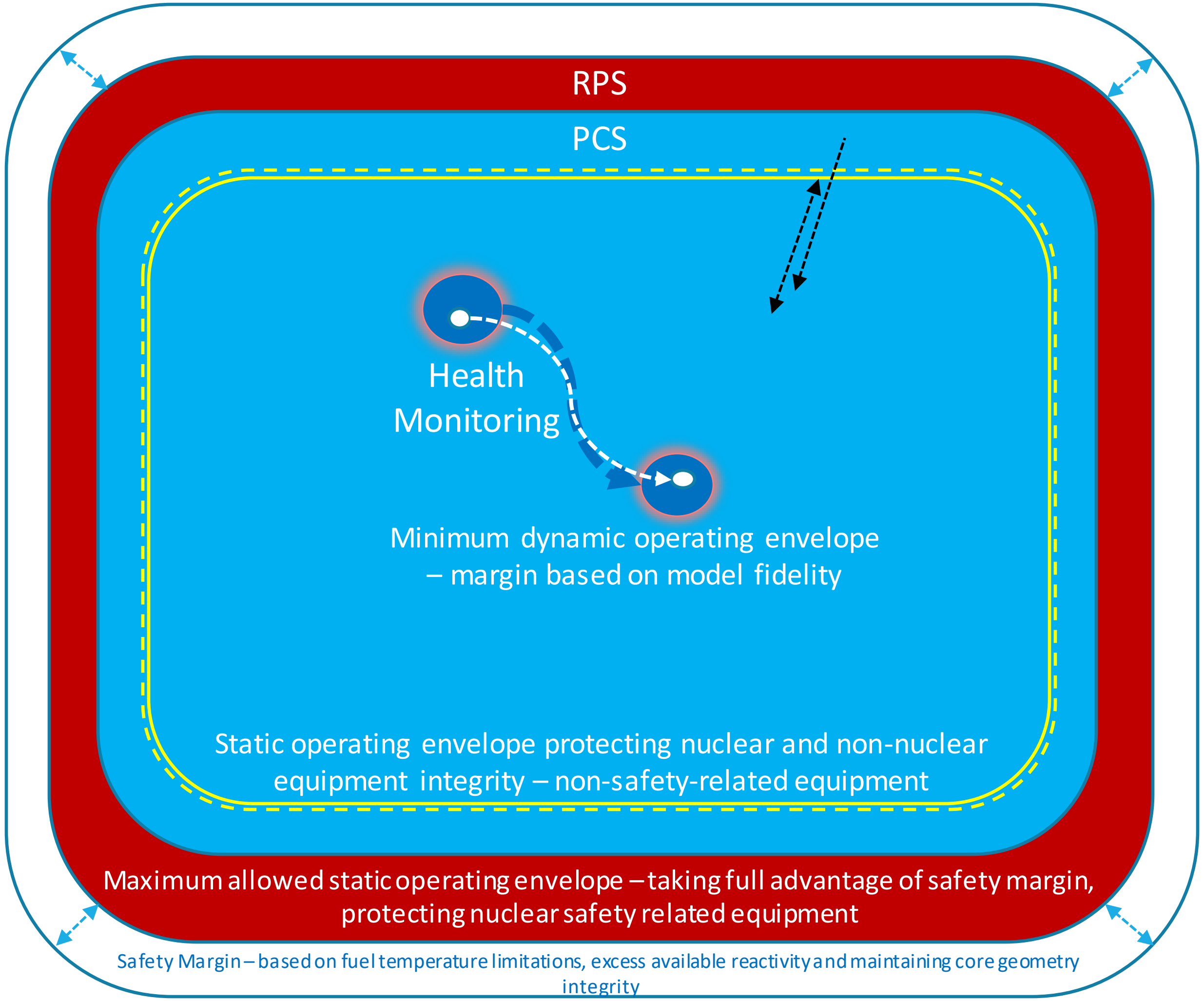
- Semi-autonomous control room (MCR)

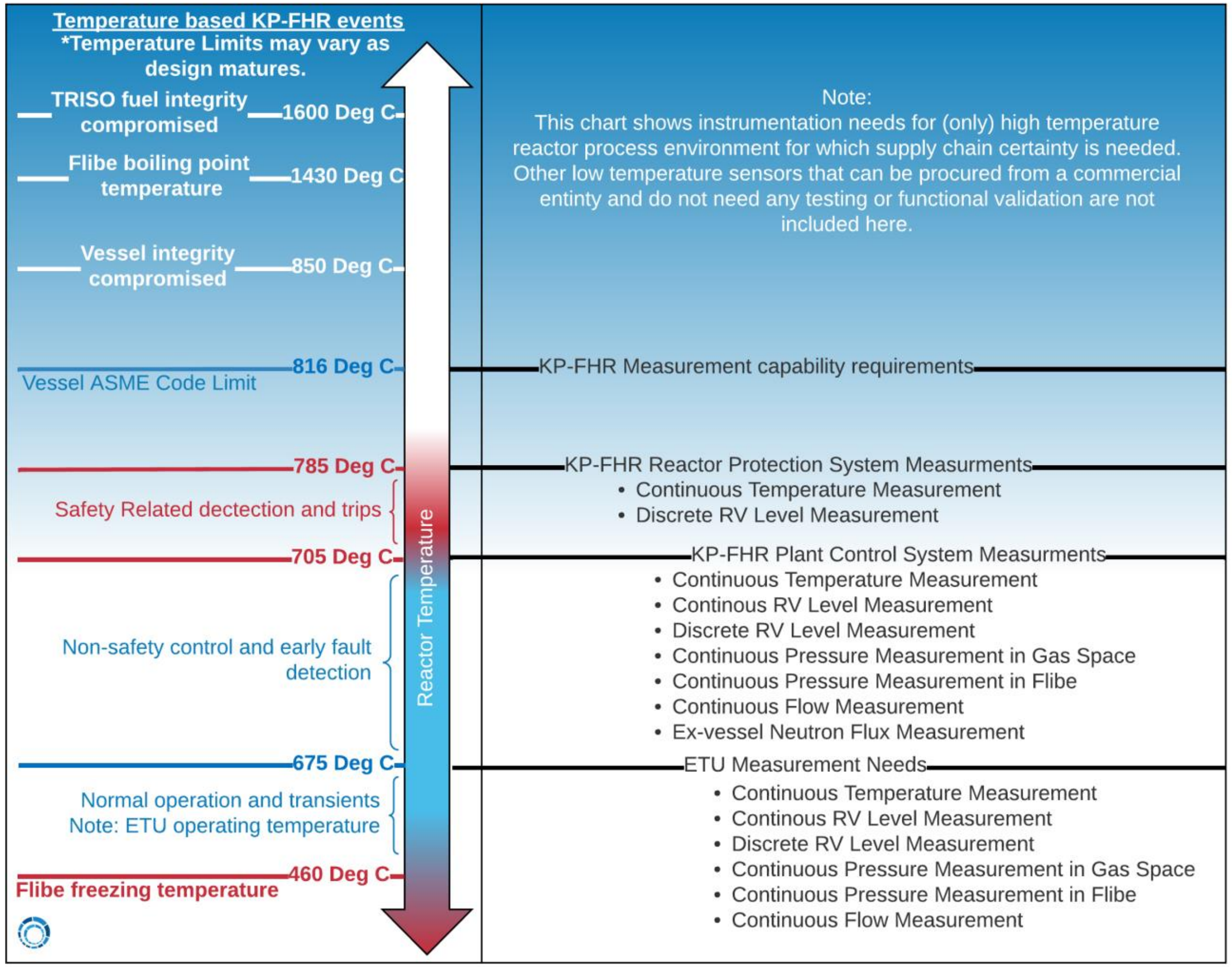
Semi-autonomous Industrial Grade HMI Technology

- Electrical supply

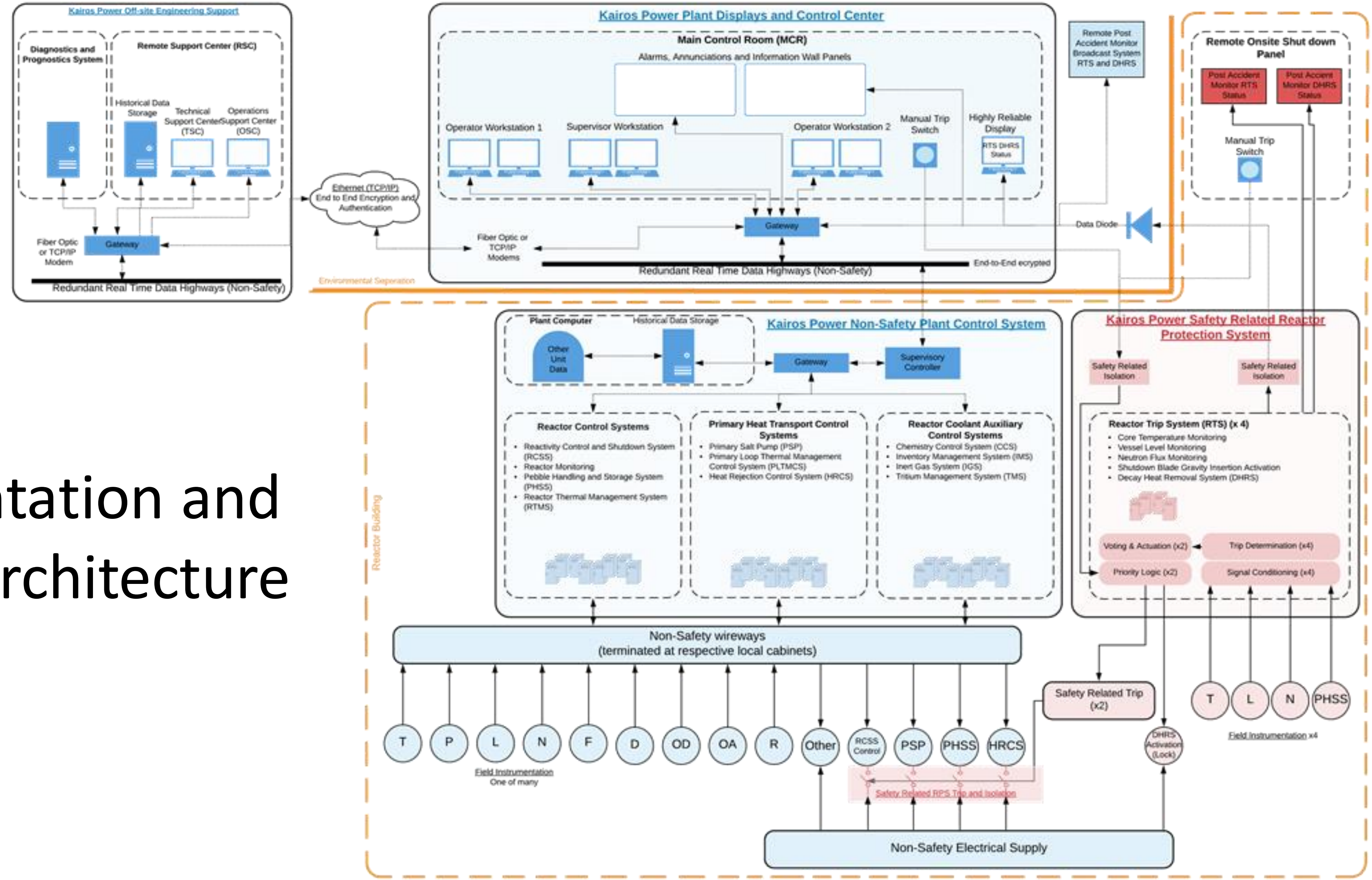
Basic Ohm Law Triangle (V = I.R)

Plant Protection, Control, and Health Monitoring Operating Envelopes

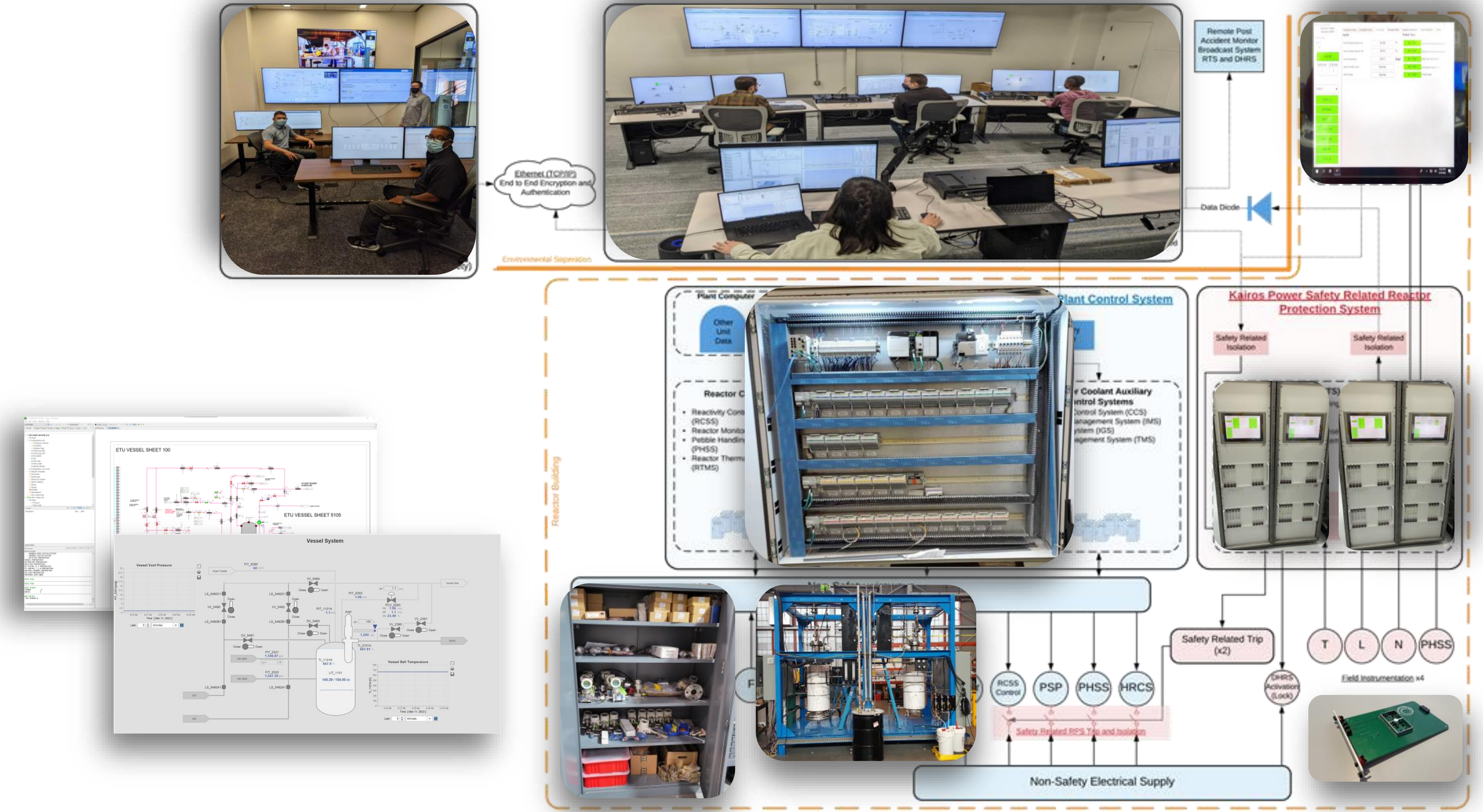




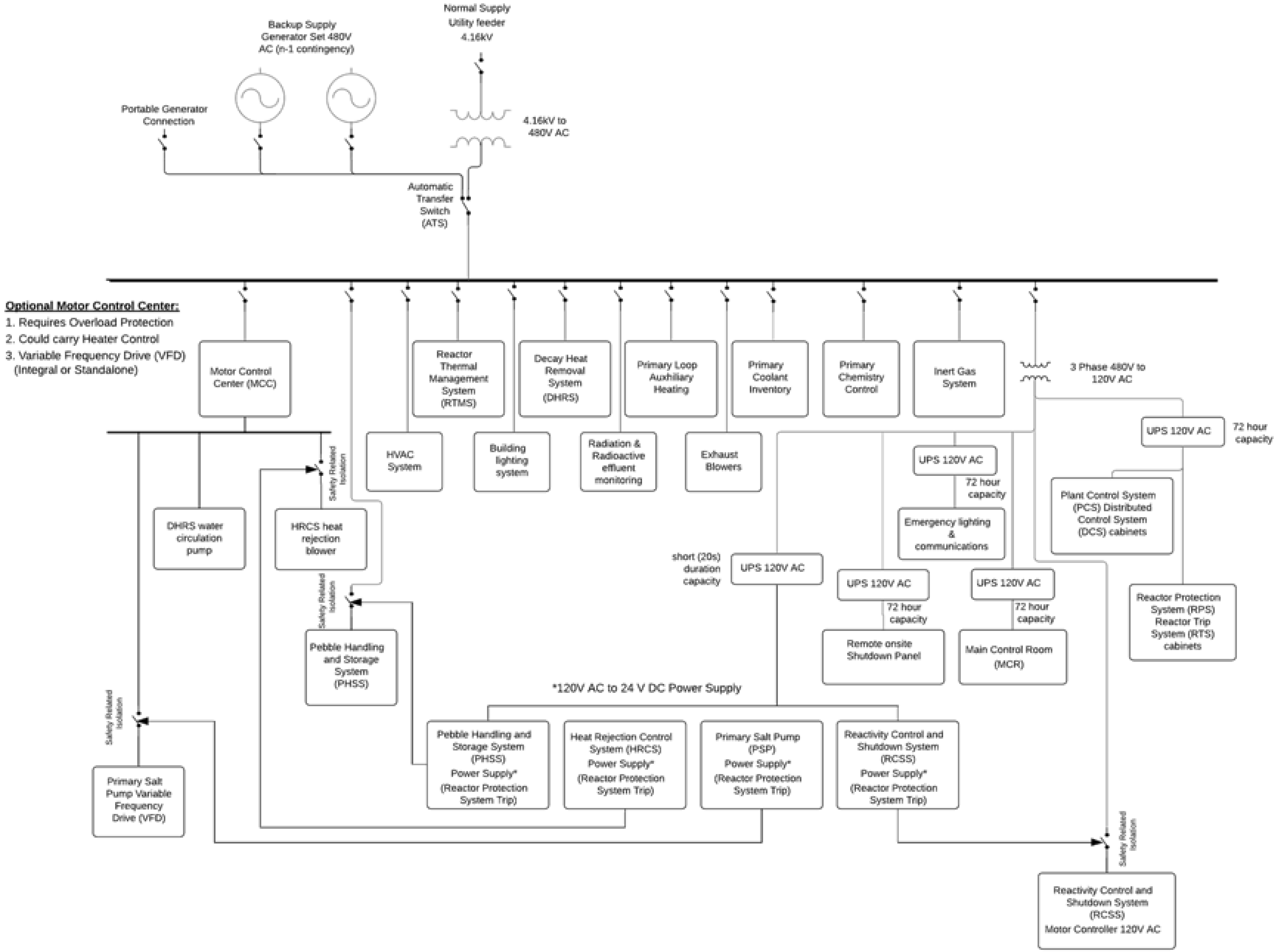
Instrumentation and Controls Architecture



Instrumentation and Controls Architecture



Electrical Architecture



Safety Case

JORDAN HAGAMAN – DIRECTOR, RELIABILITY ENGINEERING

Safety Case Approach

- Deterministic approach consistent with NUREG 1537, Chapter 13
- To demonstrate compliance with regulatory dose limits, a Maximum Hypothetical Accident (MHA) that bounds the Chapter 13 postulated events is analyzed for dose consequences
  - MHA not physical
  - MHA includes conservatisms that maximize source term
  - MHA includes a postulated release of radionuclides
- To ensure that the postulated events are bounded by the MHA:
  - List of postulated events is comprehensive to ensure that any event initiator with the potential for radiological consequences has been considered
  - Initiating events and scenarios are categorized, so that a limiting case for each group can be qualitatively described in CPA (quantitative results will be provided with OLA)
  - Acceptance criteria are provided for the important figures of merit in each postulated event group to ensure the potential consequences of that event group remain bounded by the MHA as the design progresses
  - Prevention of an event initiator is justified in PSAR

List of Events Postulated

- MHA – Hypothetical heat-up with conservative radionuclide transport
- Insertion of Excess Reactivity
- Salt Spills
- Loss of Forced Circulation
- Mishandling or Malfunction of Pebble Handling and Storage Systems
- Radioactive Release from a Subsystem or Component
- General challenges to Normal Operation
- Internal and External Hazard Events

Maximum Hypothetical Accident

- Hypothetical heat-up event with conservative assumptions meant to drive radionuclide release:
  - Pre-transient diffusion of radionuclides from the fuel in the reactor core is neglected
  - Prescribed hypothetical temperature histories are applied to the transient
  - The gas space is not credited for confinement of the radionuclides that release from the Flibe-free surface
  - Conservative, unfiltered, ground level releases
  - Conservative tritium modeling
  - A bounding vessel void fraction is assumed to facilitate the release of low volatility species in the vessel via bubble burst.

Location and Duration	Whole Body Dose (rem)		Thyroid Dose (rem)	
	10 CFR 100 Limit	MHA Result	10 CFR 100 Limit	MHA Result
Exclusion Area Boundary (First 2 hrs at 250m)	25	0.227	300	0.235
Low Population Zone (30 days at 800m)	25	0.059	300	0.081

Postulated Events

- The postulated event methods are provided in KP-TR-018, "Postulated Event Analysis Methodology" (incorporated by reference in PSAR Ch. 13)
- The phenomena for each postulated event group that have the potential to increase dose consequence are identified as figures of merit
- Acceptance criteria are defined for the figures of merit that will ensure that the limiting event in each postulated event group is bounded by the MHA
- Validation and detailed final analyses of the postulated event groups will be performed for the operating license application

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NRC STAFF SAFETY REVIEW OF THE KAIROS HERMES TESTING FACILITY CONSTRUCTION PERMIT APPLICATION

Presentation to the Advisory Committee on Reactor Safeguards

Thursday, April 21, 2022

Edward Helvenston, Project Manager
Non-Power Production and Utilization Facility Licensing Branch,
Office of Nuclear Reactor Regulation, U.S. NRC

Background

- The Kairos Hermes CP application, and other similar applications, represent a significant, watershed moment for nuclear energy and technology in the United States.
- Kairos and other designers and operators of new reactor technologies must demonstrate safety; however, for its mission of independently reviewing licensing applications for reasonable assurance of adequate protection of public health and safety, the NRC staff is committed to performing in an effective and efficient manner.
- The NRC staff's review focuses on matters that are most safety significant, and the scope of the review is commensurate with the risk posed by the designs.
- This type of review requires innovative and novel approaches.

Responsibilities and Coordination

- The Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU), in the Office of Nuclear Reactor Regulation (NRR), has primary responsibility for licensing activities for testing facilities licensed under 10 CFR Part 50, including initial licensing of non-power reactors using advanced reactor technologies.
- Hermes review is using a core team approach
  - Core team includes PMs and technical reviewers from DANU, and attorney from OGC
  - One lead PM and two supporting PMs, including one non-power reactor PM
  - Core team example topics: thermal analysis; structural analysis; fuels; source term; health physics
  - Non-core subject matter expert (SME) example topics: human factors; quality assurance; fire protection; geology/seismic; emergency planning

Non-Power Reactor Licensing

- Non-power reactors licensed under 10 CFR Part 50 may be licensed as commercial facilities under Section 103 of the Atomic Energy Act of 1954, as amended (the Act), or as research and development facilities under Section 104c of the Act.
- In its CP application, Kairos states that it expects to apply for a Class 104c license, pursuant to 10 CFR 50.21(c), for a utilization facility useful in the conduct of research and development activities of the types specified in the Act.
- Therefore, the NRC staff is conducting its CP review consistent with Section 104c of the Act, which states:
  - "The Commission is directed to impose only such ***minimum amount of regulation*** of the licensee as the Commission finds will permit the Commission to fulfill its obligations under this Act to promote the common defense and security and to protect the health and safety of the public and will permit the conduct of widespread and diverse research and development."

Non-Power Reactor Licensing (con't)

- Non-power reactor types defined in NRC regulations include research reactors and testing facilities
- Per 10 CFR Part 50 definitions, "*Testing facility* means a nuclear reactor which is of a type described in [10 CFR 50.21(c)] and for which an application has been filed for a license authorizing operation at:

  (1) A thermal power level in excess of 10 megawatts; or

  (2) A thermal power in excess of 1 megawatt, if the reactor is to contain:

  (i) A circulating loop through the core in which the applicant proposes to conduct fuel experiments;

  (ii) A liquid fuel loading; or

  (iii) An experimental facility in the core in excess of 16 square inches in cross-section."
- Many 10 CFR Part 50 requirements are for power reactors and do not apply to non-power research reactors and testing facilities
- Testing facilities are subject to the requirements of 10 CFR Part 100, "Reactor Site Criteria"
- Testing facilities are subject to a few 10 CFR Part 50 requirements that do not apply to research reactors, including Advisory Committee on Reactor Safeguards (ACRS) review, and mandatory hearings for CP applications (10 CFR 50.58)

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Testing Facility Licensing Process

- Similar review process for CP and operating license (OL) applications:
  - Acceptance and docketing review
  - Parallel safety and environmental reviews
    - CP: preparation of safety evaluation report (SER) and environmental impact statement (EIS)
    - OL: preparation of SER and EIS supplement
  - ACRS review
  - Hearing(s)
    - CP: mandatory hearing on sufficiency of staff safety and environmental reviews
    - CP and OL: potential for contested hearing(s)
  - Decision to grant or deny permit or license

Risk-Informed Review

- For its CP application review, the staff's review depth and scope will be commensurate with the risk or safety significance of items under review, and consistent with the "minimum amount of regulation" requirement in AEA Section 104c
- The staff will maintain a "big picture" safety perspective of the Hermes design. The staff will tailor the scope and level of detail of the review based on the small size of Hermes and anticipated strong safety case with low radiological consequences, and as appropriate for a testing facility CP application.
- The staff's review is also tailored to the unique and novel technology described in the CP application, using the appropriate regulatory guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors." Other guidance (e.g., regulatory guides and industry standards) and engineering judgement are also used, as appropriate.

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NUREG-1537 Review Areas/Chapters

1. The Facility/Introduction
2. Site Characteristics
3. Design of Structures, Systems, and Components
4. Facility Description
5. Coolant Systems
6. Engineered Safety Features
7. Instrumentation and Control
8. Electrical Power Systems
9. Auxiliary Systems
10. Experimental Facilities
11. Radiation Protection and Waste Management
12. Conduct of Operations
    - Emergency Planning
    - Physical Security
    - Operator Licensing
    - Startup Plan
    - Human Factors
    - Quality Assurance
13. Accident Analysis
14. Technical Specifications
15. Financial Qualifications
16. Other License Considerations
17. Decommissioning
18. Uranium Conversions
19. Environmental Review

(Note that some NUREG-1537 chapters are not applicable to the Hermes CP application)

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Construction Permits

- Safety reviews for CP and OL applications are conducted in accordance with the Commission's regulations
- The level of detail needed in a CP application and associated NRC staff SER are different than for an OL (or combined operating license (COL))
  - The CP application describes the preliminary design of the facility, while an OL application should describe the final design of the facility, as well as plans and programs not provided in the CP application
- The staff must make the following findings to issue a CP, based on 10 CFR 50.35:
  - Facility has been described, including the principal architectural and engineering criteria for the design
  - Further technical or design information may be reasonably left for later consideration in the final safety analysis report (i.e., OL application)
  - Safety features or components requiring research and development have been identified
  - Safety questions will be resolved prior to the completion of construction and the proposed facility can be constructed with undue risk to the health and safety of the public
- Staff's conclusions are also based on the considerations in 10 CFR 50.40 and 50.50

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Hermes Review Schedule

- Robust and effective pre-application engagement in order to optimize safety review (e.g., regulatory engagement plans, public meetings, topical reports, and pre-application audits
- 21-month review schedule (exclusive of mandatory hearing):

Milestone	(Estimated) Completion
Application Accepted	November 2021
Draft SER with Open Items	March 2022
SER Completion (all chapters)	(November 2022)
Approved SER to ACRS (all chapters)	(May 2023)
ACRS Letter	(September 2023)

- Staff are currently conducting audits to support SER completion, including for structural design; effluents; decay heat removal system; accident analyses; instrumentation and controls; and site characteristics. Staff are also preparing audits and possible requests for additional information (RAIs) on other topics.

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