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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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KAIROS POWER LICENSING SUBCOMMITTEE

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THURSDAY

MARCH 23, 2023

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The Subcommittee met via Teleconference, at 8:30 a.m. EDT, David A. Petti, Chair, presiding.

COMMITTEE MEMBERS:

DAVID A. PETTI, Chair

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

WALTER L. KIRCHNER, Member

JOY L. REMPE, Member

ACRS CONSULTANTS:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

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

8:30 a.m.

CHAIR PETTI: Welcome, everyone. 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: Charles Brown, Jose March-Leuba -- nope, Jose is not here; sorry -- Joy Rempe, Ron Ballinger, Walt Kirchner, Vicki Bier, and Greg Halnon.

ACRS Consultants Dennis Bley and Steve Schultz are also present.

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

During today's meeting, the Subcommittee will continue our review on the staff's safety evaluation of the Kairos Hermes Non-Power Reactor Preliminary Safety Analysis. The Subcommittee will hear presentations by, and hold discussions with, the NRC staff, Kairos Power representatives, and other interested persons regarding this matter.

Part of 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 USC 552b(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 the agenda for this meeting were posted there.

We've received no written 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.

A transcript of the meeting is being and will be made available.

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

When addressing the Subcommittee, participants should, first, identify themselves and speak with sufficient clarity and volume, so that they may be readily heard.

When not speaking, we request that participants mute their computer microphone or phone by pressing *6.

So, before we start the meeting, I wanted to tell members sort of how it's going to go. We're going to hear different sections. Unfortunately, it's not whole chapters by whole chapters, trying to accommodate different people's time constraints and schedule constraints.

But, then, we will have our time to talk about our memos. We're not going to edit the memos here. We'll just have the lead members or their designee go through the memo.

I will, early next week, go through -- because I've already read them and just found some small, tiny nits -- and I will send them back to the authors, and then, hopefully, be able to finalize them. And then, they end up having to go through -- we have to sign them through the system. So, that's sort of the plan.

With that, let me ask Ed Helvenston from NRR to start us off.

MR. HELVENSTON: Yes, thank you and good morning.

I'm Ed Helvenston. I'm one of the NRC Project Managers for the Hermes review in the Division of Advanced Reactors and Non-Power Production and Utilization Facilities in the NRC's Office of Nuclear Reactor Regulation.

Staff briefed you three weeks ago on its review strategy for the Hermes construction permit application. The staff looks forward to presenting its review to the Subcommittee in today's and subsequent meetings.

And to start off today, I would like to emphasize a few points from the previous briefing.

Although the application provided only the preliminary design of a testing facility, the mission of staff is unchanged. We must have reasonable assurance of adequate protection of public health and safety.

Review guidance for testing facilities does not differentiate between the level of detail needed for a construction permit versus an operating license application or provide specific guidance on what may be deferred to the license application.

In making its determination on the types of things that may be reasonably deferred versus what is required for a construction permit, the staff used its technical judgment and, also, considered the requirements in 10 CFR 50.34(a) and 50.34(b) regarding information that must be included in Preliminary versus in Final Safety Analysis Reports.

As stated in 10 CFR 50.35, not all safety questions need to be resolved for the issuance of a permit, but an applicant is required to identify research and development which is to be completed prior to the completion of construction in order to resolve these questions.

The staff used NUREG-1537, which is the licensing guidance for non-power reactors, to perform its review. The review depth and scope were commensurate with the safety significance of areas being reviewed -- considering the small size of Hermes, the short operating life, and the safety case with low radiological consequences.

We are glad to present our review to you and look forward to your feedback and recommendations.

MEMBER HALNON: Ed, given that, I just noticed in the SER, the Draft SER, there was a lot of places where you referred to, "Yes, we'll see that in the operating license application." And there's some places that seemed like we could have said that. Is that just to kind of generalize we're going to get more detail or are they really just held to where you said --

MR. HELVENSTON: I think it's not just limited to where we said it. There might be additional things we needed to know as well. I think where we said that in the SE, it was generally because there was a particular point that we wanted to make or something we thought that was particularly important we wanted to emphasize.

MEMBER HALNON: Okay.

MR. HELVENSTON: But, no, it's not an exhaustive --

MEMBER HALNON: Okay. Great. Thank you.

MEMBER REMPE: I appreciate your remarks this way to clarify how you did this review. And also, I really appreciate you providing Appendix A of the SE to us because it helps me with what I did with Chapter 10 a couple of weeks ago, whenever it was.

I'm just thinking about in the future, and part of my thoughts are because of some interaction we had with some folks in Finland. And I'm just wondering, do you think that at some point that there needs to be more guidance on what's done? Or are you thinking, aw, just let it go? Because I can remember with the SHINE they made some significant changes in their processes when they saw what needed to be addressed. And maybe this is the way to go, is just have that flexibility. Or do you think more guidance or structure is needed to decide on how much you need to know for a construction permit?

MR. HELVENSTON: I think the flexibility is good, and having the staff and the Applicant both be able to use their judgment in terms of what's needed now versus what can be deferred. I know we don't have any specific guidance at this point, and NUREG-1537 does not distinguish between the CP versus the OL. But, at this point, you know, I do think that, as we do these reviews, the SHINE and Northwest Medical Isotopes, the Hermes review -- you know, we have a construction permit application for Abilene Christian University now -- I think we are getting some experience and establishing some precedent in terms of what sorts of things we really do need in the CP and what we can defer.

I'm not aware of any plans to develop specific guidance at this time. You know, there is a NUREG-1537 revision ongoing. That is something that it may be worthwhile for the staff to think about in terms of something that would supplement or clarify some of what's in the guidance, you know, to give some insights on what we need in the CP versus the OL.

MEMBER REMPE: I really like Appendix A and that option, really having a place where they can find everything.

Anyway, thank you.

MR. HELVENSTON: I think, with that, I'll turn it over to Kairos to present Chapter 1.

MR. PEEBLES: All right. Thank you, Ed.

This is Drew Peebles. I'm a Senior Licensing Manager at Kairos Power.

Can you hear me okay?

CHAIR PETTI: Yes, we can hear you.

MR. PEEBLES: Okay. While Rachel is bringing up the slides, I wanted to thank the ACRS for the opportunity to present this overview of the Hermes PSAR, as well as the Committee's reviews of previously submitted Kairos Topical Reports.

As you know, we've been very active in pre-application engagement. So, at this point, we have NRC approval of 10 Topical Reports that we've -- I'm sorry -- 11 Topical Reports that we've submitted in pre-application space. And we look forward to further engagement as we continue through the licensing process.

Sorry, we're having technical difficulties with the slides.

(Pause.)

Sorry, there's a bit of a delay on the connection. We're trying to bring up the slides.

(Pause.)

Oh, there it is. Okay.

So, as I said, my name is Drew Peebles. I'm the Senior Licensing Manager at Kairos Power.

Next slide, please. You've seen this slide in all of our presentations to the Commission, but we are a very mission-focused company. So, we like to begin every presentation reiterating our mission statement, 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."

And we firmly believe we can't achieve this mission unless we develop a technology that is both affordable and, most importantly, safe.

So, just a little bit about Kairos Power. I mentioned our mission statement on the previous slide, and we intend to meet that mission by deploying our fluoride-cooled, high-temperature reactor that we refer to as a KP-FHR.

We're based in Alameda, California with several locations around the country, including a large manufacturing facility in Albuquerque, New Mexico. We have over 300 full-time employees, and those are mostly engineering-focused positions

We have aggressive cost and schedule targets for deploying the KP-FHR in order to provide a clean energy alternative the number of gas plants that are set to retire in the 2030s.

So now, I'll move on to the reason that we're here, our non-power reactor Hermes. Kairos is following the two-step licensing pathway provided in 10 CFR 50 for Hermes. We submitted the construction permit application in September of 2021, consisting of the Environmental Report and the subject of today, a Preliminary Safety Analysis Report, or PSAR.

It's worth mentioning that the next step in the licensing process will be the operating license application, which will include the Final Safety Analysis Report, which will contain design and safety information with much more finality than you will see in the PSAR.

We used the Non-power Reactor Standard Review Plan, NUREG-1537, to format our application, and there are a few chapters in that format that don't really apply to Hermes, which I'll touch on in the next slide.

But, as Ed mentioned before, the Standard Review Plan doesn't always differentiate between content that's required for the PSAR versus content that will be required for the FSAR. However, 10 CFR 50.34(a) does have a list of what is explicitly required to be in the PSAR.

I'm mentioning this upfront because there may be several areas that you might be interested in more detail across the application that we will not have until the operating license application phase of the licensing process.

I've included the text from 10 CFR 50.35, which states that the NRC will not be approving the safety of any SSC at the PSAR stage, unless specifically requested by the Applicant. Kairos did not request safety approval of any item in the PSAR. So, that should be kept in mind when judging the preliminary safety and design information that you'll see presented throughout the next few weeks.

Next slide, please. So as I mentioned on the previous slide, we used the Non-Power Standard Review Plan to format our application. And this slide shows the titles of the chapters, which are consistent with that NUREG. Most of the titles are self-explanatory. So, I won't read through them all, but I will point out that the chapters with an asterisk on the side are not applicable to Hermes.

There's no Chapter 10 content because, although this is a test reactor, the testing is a demonstration of KP-FHR technology covered by the system design and analysis envelopes covered in the other chapters, not separate experimental facilities.

There is no content in Chapters 16 or 17 or 18 because we are not asking for a license on any of those items.

I also pointed out several of the chapters that have minimal content. I probably should have included Chapter 11 in that list as well. These chapters apply to Hermes, but there is little information required at the construction permit application phase. For example, the Radiation Protection Program, most of the programmatic elements aren't explicitly required in 50.34(a) for the PSAR, but they are required for 50.34(b) when we submit the FSAR.

Next slide, please. So I mentioned the Topical Reports that we have submitted in pre-application space. Before I talk to this slide, I want to point out an error. KP-TR-007, "Quality Assurance Plan," is not referenced in the PSAR. That Quality Assurance Plan was developed for the commercial reactor, based on an NQA-1 program. We are using an ANS 15.8 QA program for Hermes, which we'll discuss in more detail during the Chapter 12 presentation.

So, as I mentioned, all of these Topical Reports are approved. They all have Final Safety Evaluation Reports issued. We leverage these heavily throughout the application.

We have two Technical Reports, "Core Design Methods" and "Postulated Event Methodology," that are referenced in Chapters 4 and 13, respectively.

Next slide, please. So I only have two slides for an overview of Chapter 1, because it's mostly a summary chapter with pointers to the rest of the chapters in the PSAR. So, a lot of the detail on any of the items in Chapter 1 will be discussed in the presentations over the next few weeks.

So, starting off, the purpose of Hermes is to test and demonstrate the key technologies, design features, and safety functions of KP-FHR technology. And as I mentioned, for not having much content in Chapter 10, that is why -- we're testing and demonstrating the technology design features and safety functions, not any external experiments or anything that we need to clarify in Chapter 10.

It's a 35-megawatt, thermal, non-power reactor facility and we're licensing it for a four-year lifetime. The operating parameters will be discussed a little more in the Chapter 4 presentations.

The location that we've selected is the Oak Ridge, Tennessee, East Tennessee Technology Park, which is the former site of the Oak Ridge Gaseous Diffusion Plant. And we'll talk a little bit more about that later today in the Chapter 2 presentation.

The principal design criteria for Hermes are based on the principal design criteria we have in our approved Topical Report KP-TR-003. And those principal design criteria are based on the advanced reactor design criteria in Reg Guide 1.232. And I'll talk a little bit more about those in the Chapter 3 presentation after this one.

We have low consequences from this facility due to the inherit safety features. I pointed out two major ones there: the robust fuel design and the flibe coolant. And I'll talk more about the functional containment strategy in both the Chapter 3 and Chapter 6 presentations.

Our engineered safety features that we describe in Chapter 6 are the functional containment strategies as well as the passive Decay Heat Removal System that we call the DHRS. And we'll discuss that in more detail during the Chapter 6 presentation.

Our Instrumentation and Control System monitors and controls plant operations, and that will be discussed during the Chapter 7 presentation.

Our non-safety-related electrical system provides the normal and backup power to the facility, and that will be discussed during the Chapter 8 discussion.

All of our auxiliary systems are not-safety-related, including things like the chemistry control, inert gas, and tritium management systems.

Those will be discussed in more detail in the Chapter 9 discussion.

Next slide, please.

MEMBER BALLINGER: This is Ron Ballinger. I have a question. The license is for four years?

MR. PEEBLES: That's correct.

MEMBER BALLINGER: Let's say something happens and you need to go further. What's the long pole in the tent to increase the number? In other words, let's say you have to use it for six years.

MR. PEEBLES: Yes, we would have to amend the license. So, we'd have to submit another revision to the application.

MEMBER BALLINGER: Yes, I mean, I know that, but, I mean, is there something in this current application which would have to be very significant to go an extra two years?

MR. PEEBLES: I don't think we've done that delta analysis, because we really are planning a short lifetime for this reactor. It's just to demonstrate that we can produce a low-cost nuclear heat iteration or technology. So, we're not really looking to go further than that. If something did happen, we'd look into that with the effort required for a license amendment.

MEMBER BALLINGER: Thank you.

CHAIR PETTI: But, just to follow on, if experience and capacity factor was really low, lower than you anticipated, and you were at least hoping to get so many effective full-power days, that might be a reason you would want to go longer than four calendar years.

MR. PEEBLES: We fully expect the capacity factor to not be high, being a first-of-a-kind and a test reactor. I'm not sure that that would be the driver, but I don't know if I can speculate on reasons why we would extend the lifetime right now.

MEMBER REMPE: Some countries give credit for the fact that the reactor was shut down; for example, if you couldn't get the fuel and it was delayed, new fuel, for three months. The NRC, though, does not. They have a calendar date, right? And so, if you say you're going to start up on year one day one, at the end of your four 365 days, you must shut down, right? The NRC doesn't give credit for downtime.

MR. PEEBLES: That's correct.

So, on this next slide, pointing out that we have two nuclear safety classifications. It's binary. All the SSCs are safety-related or non-safety-related. We'll discuss that a little bit more in the Chapter 3 presentation after this one, including some modifications we made to the safety-related definition.

Any potential events for this facility, we call postulated events, and those are evaluated using a deterministic safety analysis with a maximum hypothetical accident to demonstrate dose compliance. And we'll discuss that a bit more in the Chapter 13 presentation.

Radioactive waste management and radiation protection, I discuss that on one of the previous slides, that there's little content, but we do have a lot of text in Chapter 11 discussing how the programs will comply with the Part 20 requirements.

Experimental capabilities. I mentioned that before, that we don't have any external experimental needs. So, the capability to perform the testing associated with the purpose of the reactor is included in normal system design, which we describe in all of the PSAR chapters. No additional facilities or capabilities are required.

We also have a list of research and development programs that Ed mentioned in 1.3.9 to resolve safety questions, and we've committed to resolving those before the completion of construction of the facility.

And then, finally, in Chapter 1, I'll mention that Hermes is a single unit. So, any of the shared systems requirements, such as the PDC for shared systems, don't apply to Hermes because it's a single unit plant.

And that concludes my material on Chapter 1. I'm happy to take any questions.

MEMBER KIRCHNER: Drew, this is Walt Kirchner. I'm sorry I'm not there in person.

On experimental capabilities, to include testing of fuel irradiation, that was something that was implicit in the Fuel Qualification TR. Do you have an estimate of what kind of burnup you might achieve? We just discussed that it may be a low capacity factor, but are you looking to do, in effect, or try to achieve a burnup that would be comparable to what you would see in the power reactor?

MR. PEEBLES: No, we think it will be pretty low. I'm looking at one of our Chapter 4 people. So, it's 6 percent, about 6 percent. It's percent (audio interference).

MEMBER KIRCHNER: How would that compare for a target burnup in an actual power plant?

MR. PEEBLES: Yes, we don't have our core design people in the room yet. So, can we get back to you on that?

MEMBER KIRCHNER: Yes. Sure. That's fine. Thank you.

MEMBER BROWN: Hey, Walt, are you talking about full life burnup when they say 6 percent? Or are you talking something smaller?

This is Charlie.

MEMBER KIRCHNER: No, I was just looking for a comparison -- because it includes fuel irradiation -- just what kind of burnup they would achieve versus what they're looking to achieve when they go to an actual power plant application.

MR. GARDNER: Walt, this is Darrell Gardner, Senior Director of Licensing with Kairos. Just a quick comment.

I think I understand the question. It's important for us communicate that Hermes isn't intended to be necessarily the sole way that we qualify fuel. We have other ways of qualifying fuel for the burnup onboard the commercial reactor.

So, while we do talk about fuel irradiation testing, there is data that we will recover from operating Hermes, but it isn't the only dataset that we need for qualifying commercial fuel.

MEMBER KIRCHNER: No, it's just that it's a great opportunity to try, you know, to qualify the actual fuel form in its final configuration; i.e., a manufactured pebble in a flibe environment.

MR. SCHMIDT: This is Jeff Schmidt from the staff.

I just wanted to give you some context. So, they're limited to 13.2 for that, based on the AGR kernel size. So, you're getting, roughly, potentially, 6 percent out of the 13.2 limit.

DR. SCHULTZ: This is Steve Schultz.

The same type of question may be asked about materials corrosion. Over the period of operation, you listed here that the capabilities will include of materials corrosion and irradiation. Are you going to speak later about the details there associated with the evaluations that will be done in that area?

MR. GARDNER: So, Darrell Gardner again.

I think the short answer is we're not planning to. The details of our materials qualification program are in the Topical Report that was approved, the High Temperature Materials Topical Report. Again, Hermes will be collecting data in a number of areas. It's not necessarily intended to be the sole vehicle to achieve the complete set of answers described in the Topical Reports for fuel qualification or materials qualification or graphite qualification.

DR. SCHULTZ: I understand that. I'm just asking about what you mean by experimental capabilities in these areas -- the fuel irradiation, materials corrosion and irradiation. What will you do to support the statement here?

MR. GARDNER: Yes, I think what we're intending to describe is operating the facility is recovering data. There's not anything special -- it's what Drew mentioned before -- there's not special experiments that are being run by the facility. It's simply operating the facility and collecting and examining components and data.

DR. SCHULTZ: And that's a good way to express it. Thank you.

MR. GARDNER: Yes, sir.

MEMBER REMPE: And also, have you had a chance to look at Appendix A of the SE? The staff actually points out many places where additional details must be provided in the OL along that track. And that's kind of where I was at when I was trying to say we need more details. And the staff, basically, gave me some confidence that they also feel that way.

DR. SCHULTZ: Good. Thank you.

MEMBER HALNON: This is Greg.

Just to finish off the four-year operating license discussion, I started thinking about what that would do. When you get issued the license, you'll be within some decommissioning windows that are already established. For instance, about five years before the end of the operating license, you're supposed to submit a decommissioning plan.

So, I would suggest really looking at 50.82 and 50.75 and make sure that this short operating window is not going to put you in a period of scrambling to get your decommissioning stuff in place, both funding and plans relative to the four years, especially since you decided not to go for a possession-only license at the end of this, which means you'll have to have a continuation of the operating license.

So, it's an interesting thing. I think make sure that you've done that homework and established the matrices and make sure you're not going to get yourself in a non-compliant situation.

MR. PEEBLES: Appreciate the comment.

Thank you.

CHAIR PETTI: Okay. Ed?

MR. HELVENSTON: And with that, I think I'll go ahead and present Chapter 1, the staff's review. I'll get the slides up here. Thanks, Ben.

Yes, so next slide, please.

So, yes, I'll be presenting the staff's review of Chapter 1 following the presentation from Kairos. I'll just give a very brief introduction and overview of the regulatory requirements that we primarily used to conduct our review, as well as a very brief overview of the review.

As Drew mentioned, Chapter 1 is primarily a summary chapter. So, there aren't a lot of specific staff conclusions on Chapter 1, but I will briefly go over the findings and conclusions that we do have in that chapter as well.

Next slide, please. So as Drew already stated, Kairos has requested a construction permit for a 35-megawatt, thermal, non-power reactor facility that will be called Hermes. The purpose is to test and demonstrate key technologies, design features, and safety functions for Kairos' KP-FHR technology and its SSCs. The reactor will be located in the East Tennessee Technology Park near Oak Ridge, Tennessee.

Based on the information in the application, Hermes would be licensed as a non-power reactor under 10 CFR Part 50, under a Class 104© license, for a facility that's used for testing, research, and development.

Next slide, please. So as I mentioned, this is a summary of the primary regulations in Part 50 that we used, that are applicable, that we used to conduct our review of the Hermes CP application. Just a few of these I'll just point out.

10 CFR 50.33/50.34 lay out the information that's required to be in an application for a construction permit.

10 CFR 50.35 is the specific findings that the staff is required to make for the issuance of a construction permit that we make in our SER to support our conclusions.

10 CFR 50.40 includes common standards that are findings that the staff has to make for any type of application, whether it's a CP or an OL, or a combined operating license, or some other types.

And then, a few of these others are regulations that we considered in our review as well.

Next slide. So one thing I do want to emphasize, over the course of our review, in our audits and requests for additional information that we asked, Kairos did submit some supplements to the NRC in response to our requests, and the bulk of this information is primarily incorporated and updated in the latest revision of the PSAR, Revision 2, which was dated February 24th. And that's the document that primarily forms the basis for the findings in the NRC staff's Safety Evaluation.

In terms of the construction permit, I think, as we've mentioned, that will allow Kairos to proceed with construction based on the preliminary design information provided in the application. But, as stated in 10 CFR 50.35, and based on Kairos' request, it will not approve a final approval of the safety of any design feature or specification, unless specifically requested by the Applicant, which Kairos has not requested at this point.

In terms of the primary guidance for our review, we used NUREG-1537, which is the SRP for non-power reactors such as Hermes. We also used other guidance, such as Regulatory Guides and ANSI standards and engineering judgment, as applicable, to make the findings in the CP, as we discuss and identify in the SE.

Next slide, please. So, next, I'll just say a little bit about what we did look at in Chapter 1 in our review and some of the findings and conclusions that we made.

You know, there's a few sections that were just introductory, general material and PSAR at 1.1 and 1.3. The staff did review that information and summarize it, but we didn't make any specific findings on it.

In PSAR Section 1.2, Kairos provided some information on the overall safety case for Hermes, which we reviewed that information, and in addition to the rest of the information in the PSAR. And in our SE Section 1.2, we describe our overall findings that support our conclusion that the applicable standards and requirements of the AEA and the NRC regulations have been met for the issuance of a construction permit.

In terms of PSAR, our Section 1.4, we found, based on the fact that it's going to be, essentially, a standalone facility, and any offsite infrastructure, like offsite power, is not needed to perform a safety function, that there was no additional information that was needed what was provided for that item.

Next slide. PSAR Section 1.6, a summary

of operations, we found that, based on the information in the PSAR on how they'll operate the facility -- for example, the four-year lifespan -- we found that that's consistent with relevant assumptions and analysis later in the PSAR in which the safety implications of the proposed operations and how they'll operate the facility are evaluated.

In PSAR Section 1.7, we determined that, based on the evidence of good-faith negotiation with DOE and terms of the disposition of used fuel and high-level waste from Hermes, that they've satisfied the requirements of that Act for the issuance of a construction permit.

In PSAR Section 1.8, because it's, essentially, a new facility, there's no existing facilities or modifications, that section is not really applicable.

Next slide. So as we pointed out, the regulation in 10 CFR 50.34(a)(8) requires an applicant for a construction permit to identify SSCs requiring R&D and describe what their program for that will be. In PSAR Section 1.3.9, Kairos did identify a number of research and development activities -- fuel pebble behavior; developing a material surveillance/sampling program; qualification testing of high-temperature material; analysis of potential graphite oxidation in postulated events; validation of computer codes; the fluidic diode; thermodynamic data, and pressure correlations used in the coolant system design; process censor technology, as well as the reactor coolant chemical monitoring instrumentation. And those systems, they're listed in Chapter 1, but they're discussed in more detail in some of the later sections of the PSAR as well.

And this section, we found it consistent with Regulation 50.34(a)(8). Kairos committed that they will complete these R&D activities prior to the completion of construction, which the expected date for that is December 2026.

And as we identify in Appendix A of our SE, we do have these activities listed and tracked in there, and we'll verify that those activities get completed prior to the completion of construction.

MEMBER REMPE: I have a question. I was really glad to see you also have focused on their need to have something to monitor the level of the coolant, since flibe is considered part of their containment strategy, functional containment strategy. In your discussions with Kairos, have you thought about whether this needs to be safety-related or not, this coolant-level sensor?

MR. HELVENSTON: I think some of the sensors will be safety-related. I'm wouldn't want to speak to that specific one.

I don't know if you have any information further on that, Jeff.

MR. SCHMIDT: Yes. This is Jeff Schmidt from the staff.

Yes, I would expect that to be safety-related because it forms part of their functional containment. Plus, it also maintains coolant to the pebble and TRISO. So, yes.

MEMBER REMPE: I'm glad to hear that because, again, that level sensor isn't even identified in their Chapter 7 PSAR table. It's something that you guys have focused on. And so, I was glad to see that -- unless it's in the revised PSAR. I didn't see it in the -- I think it's like Table 7-3 or something. I may have it wrong.

MR. HELVENSTON: I don't recall, but I know it's mentioned throughout the PSAR, the level sensor, but I don't know --

MEMBER REMPE: Yes, it's not one of the process ones that they identified. But, again, I was looking at an old one and maybe I need to check the update.

MEMBER HALNON: Ultimately, it was described, and there's other functional aspects of different systems that protect the level, like the anti-siphoning stuff. And that level of detail, you know, you want it now, but, obviously, it's just not there yet.

So, we'll be looking, definitely, at what's safety-related versus non-safety-related, not just interfaces, but some of those systems that clearly look like they are -- I mean, we all know what large light water reactors are supposed to do and how those instrumentations work, but, at least from my perspective, I'll be looking at equivalencies in what I would consider safety-related to protect the reactor core versus how it's described now. Because, right now, you just can't tell. You just want to know.

And that was the hard thing about doing this review, was you want that next step of level of detail. And that's one of my questions earlier, is there's just so much of the details that come back from the operating license. You just can't even imagine Appendix A being of everything because it would be so large, what you needed.

And we don't want to do the design for

them. So, it would be interesting to get it, you know, as soon as we can.

CHAIR PETTI: I just had a question on the anti-siphon. It's not on the list. Is there going to have to be testing to prove that?

MR. HELVENSTON: This is a list of specific research and development programs that Kairos has identified. It's not an exhaustive list of everything that they are going to need to maybe learn more about or make a final determination on before the operating license.

And I think our Appendix A maybe is even broken into like A1 and A2, where we have, you know, one of them lists the specific R&D and one is other sort of commitments to provide information on something.

MR. SCHMIDT: Again, this is Jeff Schmidt from the staff.

So, we did look at that. We didn't think it made the list for research and development because they're very simple devices that are passive, that effectively use like elevation differences and the cover gas. So, we didn't think it fell into research and development.

MEMBER KIRCHNER: May I ask what fluidic

diode device, then, does this refer to? I thought this was the anti-siphon device.

MR. SCHMIDT: Again, this is Jeff Schmidt from the staff.

No, there's two separate -- well, there's two anti-siphon devices, effectively, on the hot leg, if you want to call it that, and the cold leg. And then, the fluidic diode is what prevents, say, reverse flow going into our normal operations, but allows flow when you have the decay heat removal system in operation. So, you have lost your primary heat transport, and the fluidic diode allows flow in one direction to remove decay heat. They're separate systems.

MEMBER HALNON: It's key to natural circulation.

MR. SCHMIDT: Yes, that's right.

MEMBER HALNON: And that's the important part. The fluidic diode is very essential to the natural circulation.

MR. HELVENSTON: Next slide. So I think my last slide on this chapter, just one other section I wanted to mention is, in PSAR Section 1.5, Kairos did identify a number of reactors that have operated in the past; that, although there's never been a reactor exactly like this one, you know, there is operating experience with a number of the technologies -- the salt coolant, the TRISO, and the graphite moderator -- that are similar to what's being planned for Hermes.

We did review this information, in conjunction with what's in the remainder of the PSAR, you know, in accordance with the NUREG-1537 acceptance criteria. We did find that, based on what they provided, they've compared the design basis and safety considerations of Hermes with similar facilities, and there is some expectation that some of these aspects can perform in a similar manner, due to these comparable features, as well as that the test data and experience from these has been appropriately applied in the design of Hermes, as practicable.

I think that was all I had. So, I'm happy to take any further questions.

MEMBER REMPE: Could I explore what Greg brought up about this five-year limit for decommissioning, and has that come into your review process?

MR. HELVENSTON: So, I'll say that, at the CP stage, looking at decommissioning is, typically, more of an operating license application. I think they will need to submit a decommissioning report with the OL that will have some information on kind of the timelines and the funding and areas like that. So, that's something that we'll probably take a closer look at at OL, to make sure that those things are in place and that all that is going to work out in terms of the timing of the requirements.

I'd have to check 50.82, but there may be a requirement -- the timeframes, I know for an RTR I believe are a little different than they are for other reactors. So, I'm not certain about the five-year requirement. I'd have to look at that.

MEMBER HALNON: Yes, I was just looking at 1537, and at least it said go look at 50.82 and 50.37.

MR. HELVENSTON: Yes.

MEMBER HALNON: And it does say about five years for a decommissioning plant submission. And Chapter 17 in this PSAR was left kind of open-ended, so we do expect more detail. But if you just gave it a four-year operating license, you're already well into that five-year window; plus, then, you get a two-year window for other things that you have to do. So, the amount of detail in that Chapter 17 is going to be beyond even what you would expect in a normal RTR which is going to operate for many years.

MR. HELVENSTON: Yes. Well, we'll probably have to consider that 1537 and the regulation were not written with a reactor that's going to operate for four years in mind.

MEMBER REMPE: And I did have the table wrong. It was identified as a safety-related sensor, but there's a lot of "we don't know what the sensor is" type of stuff. So, I stand corrected about what I said there.

CHAIR PETTI: Okay. Thank you, Ed.

MR. HELVENSTON: Thank you.

CHAIR PETTI: At this point, we should probably turn to the memo.

Okay. So, Chapter 1, just to go through, I, basically, summarize in the background. It's a test reactor; mention NUREG-1537; that the reactor uses salt flibe functional containment; relies on passive heat removal. Does not need enriched cross-cooling system.

Talk about what the key inherent safety features are, functional containment, atmospheric pressure, all reactivity coefficients being negative, except for the reflectors will be, typically, slightly positive. The vessel and other safety-related components within the seismic (audio interference) structure; shielding to minimize the occupational exposure, and a ventilation system to make sure we protect the workers.

The SER summary is, basically, the second paragraph talks about the R&D that Ed just mentioned, and I actually list the key measure areas and the staff is tracking these activities.

The relevant experience, which I was happy to see the slide on because that's something that we talk about a lot. And just note that they did that. They did a good job of looking at what had been done in the past.

Applicable standards. Discuss what they've got so far. Mention the disposition of waste, the DOE discussions, and that the staff found it sufficient.

So, I didn't identify (audio interference) but sort of a high-level summary. In fact, the only thing I will do is I'm going to mention to review; this is add 104© license. I think that would be important, but I didn't mention that. So, I've got a note to myself to do that.

Pretty straightforward.

MEMBER REMPE: Do you want to mention something about the discussion that Greg had about the decommissioning concerns or something that we may want to think about, since it is a four-year license? Or is this the right chapter to do that?

CHAIR PETTI: Well, probably it would be in Chapter 17.

MEMBER REMPE: Okay.

MEMBER HALNON: And that's my chapter, and I don't know if I point that out. So, I'll go back and revise it.

CHAIR PETTI: Yes, that's a good thing to put in on the revision.

Okay. I guess we're up to Chapter 2 and 3, sort of hybrid; 2.1 through 2.4, 3.2, and 3.3.

Kairos?

MR. PEEBLES: So, we had 3.1 and 3.6 next up on the agenda.

MR. HELVENSTON: Yes, I think we moved things around a little bit, due to some staff availability.

MR. PEEBLES: Okay.

CHAIR PETTI: As long as we cover them all.

So, that's all of Chapter 3, though, right?

MR. HELVENSTON: So, we're doing 3.1 and

3.6, and then, there will be a presentation that combines parts of 2 with parts of 3.

CHAIR PETTI: Okay. Okay. And then, we'll do the 2 and 3 memos together, because they won't be done until -- I'm just looking at our agenda; it's different.

MEMBER BIER: Yes, I would be curious to know when Chapter 2 is going to be up, so I can plan ahead maybe.

MR. PEEBLES: Chapter 2 is after this one.

MEMBER BIER: Oh, okay. Great. Thank you.

CHAIR PETTI: Okay.

MR. PEEBLES: All right. So, this first presentation is just on 3.1 and 3.6. As Ed mentioned, the rest of Chapter 3 will be discussed with the related portions of Chapter 2. 3.1 is the introduction and design criteria, and then, 3.6 is systems and components classification discussions.

Next slide, please. Oh, and this is Drew Peebles again, Senior Licensing Manager for Kairos.

So as I mentioned in the Chapter 1 presentation, we're pursuing a construction permit for Hermes reactor under the two-step process in 10 CFR 50.

We did an analysis of all of the NRC regulations in Title 10 for applicability to KP-FHRs, both a power reactor and a non-power reactor. So, we are utilizing the non-power reactor application-applicable regulations from that report. And again, that's another approved Topical, KP-TR-004, which is the regulatory analysis for a KP-FHR.

Table 3.1-1 identifies the design-related regulations from that Topical that are applicable to the Hermes test reactor. It's notable that we didn't find any specific exemptions from the regulations that we identified, but that's not to say that we didn't find some that weren't technically relevant. So, the example there is for combustible gas control.

And I'll mention another change for the safety-related definition which is a change because of the basis of the rule, not because we need a specific exemption.

The evaluated NRC Regulatory Guides for applicability to the Hermes test reactor, all of the Reg Guides in Division 1 are not explicitly applicable to research and best reactors, but we do use them to inform different areas. It's just noting that they're not explicitly required for a research and test reactor.

Divisions 2, 4, and 8 apply, and we consider those in the Hermes test reactor, as shown in Section 3.1.

Next slide, please. So for the principal design criteria for Hermes, I mentioned before that we submitted a Topical Report on the PDC for KP-FHRs. And that's a typo there. It should be KP-TR-003. And that Topical Report was also approved.

We've taken the PDC from that Topical Report and applied them to the Hermes test reactor with two departures. One, there's two that don't apply, and then, another, we changed the terminology from some of the PDC.

So, the two PDCs that we've identified as not being applicable to Hermes are PDC 5, which is the sharing SSCs, and that's not applicable because there is only one reactor and no SSCs are shared with another facility, and then, PDC 73, which deals with the interface between reactor coolant systems. That's not applicable to Hermes because we have no secondary coolant fluid.

Then, the changes in terminology, as I mentioned before, we have binary safety classification. All of the SSCs are either safety-related or not. So, the term "safety-significant" is not relevant to the Hermes classification system. So, we changed the words "safety-significant" throughout the PDC to "safety-related."

Also, anywhere the PDC mentioned "anticipated operational occurrences or accidents," those terms were replaced with "postulated events," to be consistent with the non-power reactor regulatory framework, where you don't bin the accidents according to frequency.

Next slide, please. So you can map any safety case license through the NRC to fundamental safety functions, and the Hermes safety case is no different. So, the three fundamental safety functions of preventing the release of radionuclides, removing an adequate amount of decay heat, and controlling reactivity are all satisfied by the safety case of Hermes.

So, preventing the uncontrolled release of radionuclides, we do that through our functional containment approach, which takes advantage of the primary barriers to release, which are the TRISO layers and the fuel, and the secondary barrier, which is the flibe coolant. And TRISO fuel will be discussed in the 4.2 presentation, and the flibe coolant will be discussed in the Chapter 5 discussion.

Any safety-related fluid systems that contain circulating radioactivity are designed as in Section 3, and non-safety systems that may contain circulating activity are designed as in Section 8 for applicable API standards.

In removing decay heat in the event of a postulated event, we rely on natural circulation within the vessel and the passive decay heat removal system to reject or transfer heat from the reactor core to the atmosphere.

And then, finally, controlling reactivity in the reactor core, we rely on our reactivity control and shutdown system, which we will discuss in the Chapter 4.2 discussion. And that controls reactivity during normal and postulated events.

So, as I mentioned, we have only two classifications, that being is safety-related or not safety-related. We did make a modification to the definition of safety-related. The basis for that definition was light-water-reactor-specific. So, we had to change a couple of things.

The original definition said that safety-related SSCs are those that you rely on to remain functional during and following design basis events to ensure, one, the integrity of the reactor coolant pressure boundary; two, the capability to shut down the core, maintain it in a shutdown condition, and then, the capability to prevent or mitigate the consequences of accidents.

So, the first bullet there, first, we took out the word "pressure" because our system is not pressurized. So, it's not relevant to the Hermes or any KP-FHR design. And then, we specified that not the entire reactor coolant boundary is important to the safety case. We don't have to have the integrity of the entire reactor coolant boundary in order to meet our safety metrics. We clarified that the integrity of the portions of that boundary that we rely on and the specific goal is to maintain coolant level above the active core.

And again, we didn't have to take an exemption to 50.2 because the basis of the rule was light-water-reactor-specific. So, it was not technically-relevant. We could propose our own definition to 50.2.

I think that's it on this slide.

The classifications of every SSC is shown in the PSAR Table 3.6-1.

Next slide, please. So for seismic classifications, which we'll talk a little bit more in the Chapter 3 discussions, safety-related SSCs are classified as SDC 3, in accordance with ASCE 43-19. All of the safety-related SSCs are located in the safety-related portion of the reactor building, and non-safety-related SSCs are designed to local building codes.

Quality classifications. Anything that's safety-related is considered quality-related. Anything non-safety-related is classified as not quality-related. And the quality-related SSCs conform to the requirements of the Quality Assurance Program, which is based on an ANS-15.8 standard.

And the seismic and quality classification of the SSCs are shown in PSAR Table 3.6-1.

I believe that is the least slide for 3.1 and 3.6. I'm happy to take your questions.

MEMBER KIRCHNER: Dave, this is Walt.

I'd like to go back to the slide on safety the definition of "safety-related." I'll start by observing the following: it's not a high-pressure system, but it is a low-pressure system. So, you do have a boundary to maintain, so that you don't have free access of air, for example.

Secondly, low-pressure systems leak as well as high-pressure systems, maybe not with the higher probability of leakage, but there have been low-pressure reactor systems that have developed leaks -- liquid metal systems, to be precise, which are like your system, essentially, operating at a low pressure with some kind of cover gas.

When you make the split like you have defined it here, then that suggests that the remainder of the primary coolant system boundary is not safety-related. Yet, you depend on that to prevent unmitigated or uncontrolled access of the air, for example, to the primary system and the core.

So, I would have thought you would have, for this particular reactor, which is like a prototype, that you would have made your arguments on safety classification, safety-related or not, based on the third bullet, not the first one.

Would you like to comment on that? Because I think your functional containment will satisfy the third bullet, in that your potential offsite exposures are far less than the applicable guideline exposures in 10 CFR 50.34.

MR. PEEBLES: So, you mentioned the rest of the reactor coolant boundary not being safety-related, and that is accurate. We are not relying on the boundary, for instance, for the primary heat transport system piping. Our safety case assumes that it fails and we release the entire inventory of the primary heat transport system, and we're still able to meet our very low dose metrics.

As far as mitigating air ingress, we do also accommodate air ingress in the safety case, which I think we'll discuss, then, in the Chapter 13 discussion as well.

CHAIR PETTI: I thought -- and again, I may have read this in the previous revision PSAR -- I thought it was beyond design basis. Has that changed?

MR. PEEBLES: It's within the design basis.

CHAIR PETTI: It's within the design basis?

MR. PEEBLES: Yes.

CHAIR PETTI: Okay.

MEMBER KIRCHNER: So, then, uncontrolled access of air to the reactor vessel is not a beyond design basis accident? You think that's a design basis event? Because, depending on the level that the fluidic diode and anti-siphon devices as a system leave, whatever the level is, your design objective is to keep the active core covered. But this will expose significant amounts of high-temperature graphite to air, as well as the free surface of the flibe.

MR. PEEBLES: So, it does expose some graphite, but not the portion that's needed to maintain the natural circulation flow path.

MEMBER KIRCHNER: But, then, you have a graphite-air reaction as well to deal with.

MR. PEEBLES: Again, we're not depending on that exposed graphite for it to maintain its structural integrity or anything. We have hold-down plates for the flibe-wetted graphite that maintains the natural circulation flow path.

MEMBER KIRCHNER: No, I get that part. So, then, the estimates of potential reaction of air with flibe and/or graphite will not exacerbate release of radionuclides?

MR. PEEBLES: Correct.

MEMBER KIRCHNER: I mean, or is that going to be something that's demonstrated in the R&D programs?

MR. PEEBLES: No. It's all included within the scope of the materials qualification that we'll do testing to quantify how much is oxidized. But we do, in the safety analysis, consider almost a complete failure of what is exposed.

MEMBER KIRCHNER: Okay. Well, my concern

is that this is precedent to stop -- divide up a primary coolant system in this manner, such that the vessel is safety-related and the remainder of the coolant envelope is not.

MR. GARDNER: So, this is Darrell Gardner again.

We don't disagree with that. It's a new technology, advanced reactors. Again, this definition is based entirely on the understanding of light water reactor technology. So, we went into this fully expecting we would need to make adjustments to this definition to be meaningful for our design and our safety strategy.

MEMBER KIRCHNER: No, I appreciate that. I'm just flagging it because this is a good time, I think, to have the conversation at this early juncture. Otherwise, it creates significant complications at the OL stage, if, indeed, the classification of equipment were to be changed, for whatever reason.

MEMBER BROWN: Walt, this is Charlie.

Does some of this definition apply because their decay heat removal system is relied on for almost all heat removal, once you're below -- well, it's on all the time after you exceed 10 megawatts thermal. So, is the DHRS tied up in this definition as well?

MEMBER KIRCHNER: No, it's a separate system from the cooling system.

MEMBER BROWN: I know that, but, I mean, does its performance allow this to be done? That's the only question I'm saying. You're ahead of me on some of the aspects you're talking about, but the overheating is not one of them, is what you're saying?

MR. PEEBLES: This is Drew Peebles.

So, the third bullet captures why the DHRS is safety-related and its performance. So, its capability --

MEMBER BROWN: No, I understand that. You know, I've read that already in Chapter 6 and, also, it's a discussion in Chapter 7. I was just wondering how it tied into this other new definition.

All right. I'll stop there.

MR. PEEBLES: Well, I just want to point out, we didn't change the third bullet there. So, anything that we rely on to mitigate the effects of can accident -- or excuse me -- a postulated event are considered safety-related.

MEMBER BROWN: Okay. But it's stated in the later chapters?

MR. PEEBLES: So, that third subbullet under "Safety-Related" -- I don't know if you can use the mouse there. Sorry. That bullet there, that captures everything that is relied upon to mitigate the consequences of events.

CHAIR PETTI: So, does that by the letter of the law make the flibe itself a safety system? Because it prevents a lot of stuff from happening, in terms of functional containment.

MR. PEEBLES: Essentially, it's a safety-related barrier.

CHAIR PETTI: Yes.

MR. PEEBLES: So, not unlike the flibe -- or excuse me -- the TRISO barriers, it's more of a commodity than it is a particular SSC, but, yes, it would still have the rigor to change the (audio interference).

CHAIR PETTI: I had another question on the classification. I just couldn't remember. The piping in the pebble-handling system, is that safety-related?

MR. PEEBLES: No, that's non-safety-related.

CHAIR PETTI: So, then, you assume you've got an accident where you've got an ingress with the pebbles?

MR. PEEBLES: Correct.

CHAIR PETTI: Do you have more slides?

MR. PEEBLES: Oh, no. Sorry, I was just taking questions.

CHAIR PETTI: Okay. Yes. I don't see any more here in the room. So, why don't we go to the staff?

MR. LE: Good morning.

My name is Tuan Le. I'm a Reactor Engineer in the (audio interference) Division.

Today, I will go over the staff review on the PSAR, Section 3.1 and 3.6.

CHAIR PETTI: Speak a little closer to the mic. Pull the microphone closer. It's really hard to -- you're very faint.

MR. LE: Can you hear me?

So, the Sections 3.1. and 3.6, I will go over the staff review and the validation of Section 3.1 and Section 3.6.

Next slide, please. The agenda, our role is Sections both 3.1 and 3.6. The agenda will be the same: the overview, regulatory basis, staff technical evaluation and conclusions, and regulatory findings.

Next slide, please. The overview for

Section 3.1. Section 3.1 is the design criteria. The Hermes reactor used the principal design criteria based on the approval of the Topical Report, the KP-TR-003-NP, the principal design criteria for the Kairos power fluoride, salt-cooled, high-temperature reactor.

In this Section 3.1, Kairos identified relevant regulations and PDCs for the Hermes reactor, as well as the NRC guidance considered in the design.

Next slide, please. The basis for this, the regulatory basis for this Section 3.1 is 10 CFR 50.34(a), "Preliminary Safety Analysis Report; 10 CFR 50.3, "Issuance of Construction Permits, and 10 CFR 50.43, "Common Standards."

Next, please. For the review process of this section, the staff evaluation, staff used the following guidance, initial evaluation for the Hermes design criteria: relevant parts of the NUREG-1537, and used the title for this NUREG is "Guidelines for Preparing and Reviewing the Application for the Licensing of Non-Power Reactors."

The staff also used the following guidance: Reg Guide 1.232 entitled, "Guidance for Developing Principal Design Criteria for Advanced Non-Light Reactors," Revision 0. That review included consideration of limitations and conditions for the staff SE for the Topical Report, the KP-TR-003-NP.

MEMBER HALNON: Tuan, how did that go when you looked at the existing guidance to what they provided? Is that work good or do we need to tweak any of the guidance in 1.232 or for the NUREG? Did it work well? I know there's only two exceptions, or whatever they took, that they mentioned.

But I guess my thought was, we're learning each time we'll get one of these new technology reactors in front of us. I'm just wondering if we learned anything significant that might inform some of the other reactors that we'll be doing in the future.

MR. LE: Yes, this is similar, comes back to the question of whether the chicken or the egg came first.

MEMBER HALNON: Yes.

MR. LE: We're learning through the process, using the 1537, and I would say the relevant part of 1537 we applied to this Hermes reactor review. So, there is a number guidance documents to be included --

MEMBER HALNON: Some learnings? Because I know that 1537 is -- what? -- a 1998 document, or something, was the last time it was revised. So, it's

20-plus years old.

And I think you mentioned that 1537 is in revision, or at least gathering information. I was just wondering if there is anything significant that came out of this review that we might, when the revision comes out, consider applying in the near-term reactors that we're looking at.

MR. LE: I don't see any significant -- I was just saying that they are using 1537 for review in the Hermes reactor. Some information that we can improve later on for the guidance will be more on the non-light-water reactor guidance.

MEMBER HALNON: Okay. So, no showstoppers, though? I mean, we went through the SHINE and the same thing. We found that it was relatively okay, with some exceptions. Similar, I guess, in this situation.

MR. LE: Yes.

MEMBER HALNON: Thanks.

MEMBER BALLINGER: Yes, this is Ron Ballinger.

I have to chime in here. I mean, to my mind, the key documents are 03 and 04, the TRs, which go through the design analysis and, especially, the PDCs, and how they match up and everything.

They'll learn a lot from that, from going through that exercise.

MEMBER DIMITRIJEVIC: This is Vesna Dimitrijevic.

I wanted to actually add something to this discussion. I mean, Ron, you are right, those TRs are very good guidance, but they are done for the reactor, not for the test; I mean for the power reactors. So, there is a difference there.

And in this discussion about what has changed, it is that 1537 is not risk-informed. So, as you can see, the definition of safety and the accidents. So, the Regulatory Guide 1.232, which is risk-informed, there is big exemptions which staff made here; that it is not necessary to use NEI 18-04 guide, which is from the risk-informed principles. And it's okay to use 1537, which is purely deterministic.

So, there is -- I mean, I don't know whether I call it "an exemption," but sort of like, you know, in the big step difference between what 1.232 is.

MR. LE: Shall I continue?

So, in staff evaluations for Section 3.1, the staff also considered the limitations and conditions from the staff SE for the use KP-TR-003-NP for design criteria. The limitation and condition for the following:

The key design feature for Hermes, the system, were those in the KP-TR-003-NP.

Secondly, the relevant manufacturing license scope is not applicable to Hermes.

The third one is the NEI 18-04 titled, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactors. Licensing Basis Development," Revision 1. It's not applicable to the Hermes design, the Hermes reactor. Instead, staff used the guidance of 1537.

The use of the term "safety-related," as was mentioned before, Hermes uses "safety-related" consistent with 10 CFR 50.2.

Terminology-wise, they consider safety-related use instead of the safety-significant, as indicated in NEI 18-04.

"Postulated event" is used consistently within NUREG-1537.

For the conclusion, the staff had the regulatory findings. The NRC staff found the design information was consistent with the guidance in Reg Guide 1.232 and applicable criteria for the NUREG-

1537.

The staff concludes that the information in the Hermes PSAR Section 3.1 is sufficient for issuance of a construction permit, in accordance with the 10 CFR 50.35 and 50.40.

Further information can be provisionally left for the OL, the application stage.

That is the conclusion of my Section 3.1. If there are no other questions, I will continue to Section 3.6.

MEMBER KIRCHNER: I have a question. This is Walt Kirchner.

Can we go back to safety-related? The slide on safety-related. Yes, Item 4. Yes.

Okay. So, you know, your definition, the staff's definition, in the current regulations, under safety-related structures in 50.2, includes, for design basis events, the integrity of the reactor coolant pressure boundary.

Now, we can argue whether that means high pressure or not, but, certainly, the intent is that this primary coolant boundary gives you a layer of defense-in-depth.

Now, for an LWR, one postulates that you lose that, and yet, demonstrate that you can meet the requirements of the third part of the definition, which are the dose limits found in, essentially, 50.34.

So, how do you reconcile this when this interpretation by the Applicant is applied to the primary coolant boundary? Because, again, in the case of an LWR, you postulate a double-ended guillotine break, and you, then, depend on the fuel design active systems, in the case of the large of the large PWRs, to ensure that you keep the dose below the 50.34 requirements.

MR. SCHMIDT: This is Jeff Schmidt from the staff.

So, if you look at the Chapter 13 event, salt spill --

CHAIR PETTI: A little closer.

MR. SCHMIDT: Okay. Sorry. Sorry.

If you look at the Chapter 13 event, salt spill, there is a series of breaks that will be analyzed. And I think they did a double-ended guillotine break in this, in the example problem. And it will drain down until the anti-siphon devices prevent further drain-down.

And as Drew mentioned, the exposed surfaces, including graphite, will oxidize due to the air ingress, and that is part of their analysis for the salt spill accident, which should be less bounding than the MHA. So, that air ingress is, effectively, analyzed, and the results of that air ingress are analyzed.

Does that answer your question?

MEMBER KIRCHNER: No, I -- yes, I have read that part of Chapter 13 as well, and I understand that the staff's position seems to be that they will demonstrate, the Applicant will demonstrate to you that air ingress unmitigated will not lead to dose consequences, as per the guidelines. But I just submit that that line of defense that the primary coolant boundary provides is safety-related in LWR systems. But I understand this is new technology. So, I'm not stuck in the past.

But I'm trying to understand, Jeff, at this point, the logic that says, okay, we're going to split this system and say that ex-vessel will design to what may be perceived by the public a lower quality than is required of the reactor vessel.

I'm just trying to kind of test, Jeff, the staff's logic here in accepting the Applicant's proposal as to how they're going to divide up the system.

MR. SCHMIDT: Yes, I think one thing that has to be considered -- again, this is Jeff Schmidt from the staff; sorry -- one thing that has to be considered is that, you know, there is also the flibe component, which doesn't exist in the light water reactors, right? So, maybe you're trading the primary coolant boundary system for now the flibe retention, right?

So, it's not necessarily an apples-to-apples comparison. They have a different technology which will pick up different aspects that maybe don't require the integrity of the primary coolant. So, I would just submit that as a consideration.

MEMBER KIRCHNER: Yes. So, then, Jeff, though, by logic -- you know, Dave brought this point up earlier -- then, does that make the flibe safety-related? Because, obviously, you've got limiting conditions of operation in tech specs about impurities of flibe which would be a concern for transmutation products, as well as retention of impurities may have an impact on retention of fission products in solution. And then, there's the whole question of, what does the presence of air in contact with the flibe due to its retention capabilities -- things that are part of the R&D or the materials qualification aspects of the program the Applicant has laid out?

So, that's my concern. It does create a rather new precedent. And at this point, yes, I would say, then, okay, I certainly understand from a technical standpoint the functional containment argument. So, I'm not contesting that. I'm just exploring, and this set of presentations gives us an opportunity to, you know, discuss this.

So, does the flibe, then, become, quote-unquote, "a safety-related system"?

MR. SCHMIDT: So, again, this is Jeff Schmidt from the staff.

You know, I think the short answer is, yes, it's part of -- it's just like fuel; that's a safety-related function.

MEMBER KIRCHNER: Right.

MR. SCHMIDT: It has certain characteristics for fission product retention. They're crediting flibe for certain fission product retention. So, I consider that safety-related.

CHAIR PETTI: So, basically, with the exception of the small about of release in the spill, functional containment is largely maintained, even though it's not in the vessel?

MR. SCHMIDT: That's correct.

CHAIR PETTI: I mean, that's sort of the --

MR. SCHMIDT: Right.

MEMBER KIRCHNER: Okay. Thank you.

CHAIR PETTI: Let's keep going.

MR. LE: So again, the overview of Section 3.6. 3.6 is regarding systems and components. This section describes the design basis for systems and components required to function for safe reactor operation and shutdown.

This includes Section 3.6.1. The title is "General Design Basis Information." This section is described as safety functions that are performed by safety-related SSCs.

The following are the safety-related functions of the SSCs:

The first one is preventing the uncontrollable release of radionuclides.

The second safety function is performed to remove decay heat following the successful reactor trip.

And a third safety function is control of reactivity.

Section 3.6.2, classifications of structural systems and components are described, how the SSCs are classified.

So, in this process, for each section, the staff evaluation used the following guidance for each evaluation:

Similar like to 3.3.1, the relevant parts of NUREG-1537 titled, "Guidelines for Preparing and Reviewing the Application for the Licensing of Non-Power Reactors."

Also, Reg Guide 1.129 titled, "Seismic Design Classification for Nuclear Power Plant," Revision 6.

Reg Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 2.

Also, Reg Guide 1.87, "Acceptability of the ASME Codes, Section III, Division 5, High Temperature Reactors," Revision 2.

Staff reviews cover the design basis, safety seismic qualification, declassification, design basis review of the staff evaluation, design construction code and standards for fluid system, considered with Reg Guide 1.143 as acceptable.

I will go over the high level of the safety function that the Hermes design met. They met it because the reactor vessel internal support coolable core geometry and natural circulation for the heat transfer to the decay heat removal system. The decay heat removal system operates to remove the heat from the core for at least 72 hours following the postulated event when normal cooling systems are unavailable.

Passive fuel pebbles and the coolant contain fission products. The reactivity control and shutdown system shut down the reactor and maintains reactor shutdown after an event.

And lastly is the safety-related portion of the reactor building to protect the reactor vessel and other safety-related SSCs for natural phenomena.

CHAIR PETTI: So, I had a question which may be for the Applicant. So, I've got the vessel, but the top of the vessel is non-safety-related and would be designed to a different ASME Section VIII versus the vessel, Section III. When you impose the seismic standard, is there any inconsistency there because of the different classifications from the ASME in terms of the seismic? I mean, you know, the nuclear stuff is fine during the seismic event, but the non-nuclear stuff -- and they're connected together -- is there an issue there? Or is it because it's seismic isolated and all that sort of goes away?

DR. DORON: This is Oded Doron from Kairos, Director of our Reactor Systems.

So, the entire vessel will be designed with the intent of Section III --

CHAIR PETTI: Ah, okay.

DR. DORON: -- including the head --

CHAIR PETTI: Okay.

DR. DORON: -- and nozzles up to the first connection surface, per Section III.

CHAIR PETTI: Okay. Thanks.

MR. LE: Next slide. So, in this portion, it is in the staff technical evaluation on the safety and seismic qualifications.

The safety classification, as was just mentioned before, the safety classification of the ASCs in Table 3.6-1 meets the definition requirements for 10 CFR 50.2 with one exception, the integrity of the portion of the reactor coolant boundary relied upon to maintain the coolant level above the active core.

Secondly, I will go to the seismic classification. The NRC staff finds that the safety and seismic classification conforms with the guidance of Reg Guide 1.29 because the safety classifications are assigned correctly through Seismic Classification SD 3-3, according to the ASCE 43-19 standard.

Some analysis and testing we perform in accordance with the Section 8.2 and 8.3 of the ASCE 43-19.

MEMBER HALNON: This is Greg.

Did that review include the interfacing portions of the non-safety systems, looking at the interfaces? There was a lot of non-safety systems close to or in the vicinity of, or connected to. And did your review of the seismic go into the non-safety systems to make sure that they were appropriately designed to that code as well?

MR. LE: I believe this code is relevant to the seismic qualification designed to the ASCE 43-19 as a classified safety-related portion, the component of equipment with the safety-related classification.

MEMBER HALNON: So, you only looked at the safety-related classes at this point -- the system?

MR. LE: No.

MEMBER KIRCHNER: Just a follow-on to Greg's question. This is Walt Kirchner.

My big concern here would be, again, the largest component attached to the vessel is the primary heat transport system and its piping. So, I presume that the analysis has to include, just to elaborate on Greg's question, whether those systems are -- by the way, I would note, as Dave noted earlier, it's nice you've got the whole reactor system on a seismic isolation. That clearly should reduce the loads that are seen by all the structure.

But have you looked at -- I'll give you one scenario to make my question a little more specific. Is the potential for the primary heat transport system loop to rupture the vessel examined? Because it's not safety-related in this classification system. In other words, you've got a significantly large mass attached to the reactor vessel. Is it designed to fail, so that it doesn't endanger or somehow rupture the reactor vessel and lead to a loss of flibe and uncovering of the active core?

MR. LE: I believe that the design of the Hermes reactor, it is designed to the criteria 2.2.1. They designed towards non-safety-related components -- piping, for instance -- risk. For example, if the non-safety-related piping failed, they designed such that the failing of the non-safety-related will not affect the safety-related components of the 2.2.1 criteria. And the 2.2.1 criteria, it's also they do it with the design, the non-safety-related components protecting the safety-related components. They do design to that criteria aspect.

MEMBER KIRCHNER: Well, see, this is where we're facing a new precedent. Because, typically, in the seismic analyses you look at non-safety and other components that could endanger the safety function of a safety-related system due to seismic loadings, or et cetera.

Here, you've got the primary heat transport system welded onto the reactor vessel. So, it's not a question of, for example, displacement and collision of two components. Now, you've got an integral system. So, typically, what would be done for LWRs is that the piping and the support systems for the entire primary loop would be designed to withstand a safe shutdown earthquake.

Here, if you're designing your safety-related components to seismic classification SDC 3, but you have an integral system, and the other part of the system is not designed to that, then that raises questions about whether that system could, in fact, affect the safety performance of the system it's connected to.

So, this is kind of, again, kind of an

unusual situation. I'm presuming that the PHTS is going to actually be designed as if it was SDC 3 because it's connected to the reactor vessel.

MEMBER HALNON: So, Walt, this is Greg.

There are in the auxiliary systems -- I can't speak to the PHTS -- but in the auxiliary systems there were descriptions of the interfaces and passing nearbys, and the stuff like that, where the seismic was addressed as SDC 3, as appropriate. I did not go through that in detail to make sure that every connection and locale -- because there's really no spatial other than, you know, the conceptual descriptions.

So, I know that, at least in the auxiliary system, there is that consideration. But my question, again, was more generalized, but yours is very specific, but it's the same question. It is, did you trace that through to make sure that the seismic design from the connected systems, nearby systems, and others, is appropriate to the safety class that needs to be done to protect the safety-related system?

So, I know that it's in there. I just don't know if it was all comprehensively looked at.

MR. HELVENSTON: Yes, this is Ed Helvenston from the staff.

If it's helpful for clarification, you know, the PDCs of Hermes, they apply to the non-safety systems in some cases, as well as the safety systems. And this may be discussed in some of the other PSAR chapters, but there is discussion of how some of those non-safety systems, such as the PHTS, meet associated SSC -- or associated PDCs. And there are PDCs to the effect of, you know, that non-safety systems have to be designed such that their operation, you know, or anything that could happen to them, won't interfere with a safety-related system.

MEMBER KIRCHNER: Yes, well, that's standard practice, of course. It's just that we have a rather unique situation here.

Well, I'm not trying to help the Applicant here, but I will. I mean, the penetrations of the auxiliary systems that Greg mentioned, by and large, are on the upper reactor vessel head, so that a break in any of those systems doesn't drain the flibe from the system below the top of the active core.

My specific concern is how the PHTS system, which is significantly larger than all those other support systems, like the Argonne cover gas, and so on, is designed seismically, so that it doesn't threaten the integrity of the reactor vessel.

MR. SCHMIDT: This is Jeff Schmidt from the staff.

So, I used that same non-safety-related to safety-related criteria for the primary heat transport system as well. So, it can't endanger the safety-related components of the vessel. We don't know how that's designed at this point, but the assumption is that that failure does not lead to the failure of, say, the vessel, such that the flibe is kept above the active fuel.

I guess that will be an area, obviously, we'll look at during the OL, when we have specific design details. So, that failure cannot lead to the failure of the vessel.

MEMBER KIRCHNER: Right, right. I'm just flagging it now, Jeff. There, obviously, isn't enough -- or I shouldn't say, "obviously." But I didn't find in the PSAR enough detail to know what the design strategy is on this particular matter, but it certainly would be a concern at the OL stage to look at this. Thank you very much.

MR. LE: Okay. Next slide. Yes.

I want to go to quality requirements. The staff evaluation in this area, that all the safety-related SSCs are assigned to the quality-related classification. This conforms to Reg Guide 1.29 guidance and is acceptable to the NRC staff.

ASME Code, Section III, Division 5, would be used to design and fabricate the safety-related mechanical components. And the ANSI/ANS-15.8, standard 1995 edition, the Quality Assurance Program used in the earlier reactors a quality program rather than the NQA-1.

The methodology of loading flow combinations conform to Reg Guide 1.143, Table 3, "Design Loading Combinations." The design trends in loading and number of cycles will be included in the OL applications.

Staff finds that the non-condensable gas in the reactor coolant system would not cause loss of function, and then, undue risk because the design already includes inert gas (audio interference) coolant.

Next slide, please. For the conclusion, the staff had the regulatory findings. Staff finds that the preliminary design information is consistent with the applicable criteria in NUREG-1537.

Staff concludes that the information on the Hermes PSAR Section 3.6 is sufficient for the assurance of the construction permit, in accordance with 10 CFR 50.35 and 50.40. And further information can be reasonably left for the OL application process.

Now, I will take any questions for Section 3.6.

CHAIR PETTI: Comments, Members?

Okay. I'm not hearing any. We're due for a break. Let us recess until 40 minutes -- oh, I have a hand up. Yes, Kairos?

MR. PEEBLES: Hi. This is Drew Peebles.

I just wanted to address some conversation that was a little back and forth on whether the PHTS was committing to a requirement to not affect the vessel during a seismic event. So, I wanted to point everyone to Section 5.1.3 in the PSAR. I'll just read it quickly.

"The design of the non-safety-related PHTS is such that a failure of components of the PHTS does not affect the performance of safety-related SSCs due to a design basis earthquake. In addition to protective barriers, the PHTS pipe connections to the reactor vessel nozzles have sufficiently small wall thickness, such that if loaded beyond elastic limits, inelastic response occurs in the PHTS piping which is non-safety-related. These features, along with the seismic design described in Section 3.5 of the PSAR, demonstrate conformance with the requirements of PDC 2."

So, I wanted to point that out because it sounded like we weren't sure if that was in the application or not.

MEMBER KIRCHNER: Drew, thank you for pointing that out. This is Walt Kirchner.

Yes, I saw that. And so, it sounded like, to me, the design philosophy is that you'll intentionally take a break there.

MR. PEEBLES: Correct.

MEMBER KIRCHNER: Which is an interesting design approach. What does that mean, then, for the support of the rest of the structure in terms of the rest of the loop then? I know this is Chapter 5. We're getting ahead of tomorrow. But why don't you just save that for tomorrow and I'll ask the question again?

MR. PEEBLES: Okay.

MR. GARDNER: This is Darrell Gardner.

CHAIR PETTI: Actually, we're not doing Chapter 5 until April 4th, I believe.

MR. GARDNER: This is Darrell Gardner. I wanted to add one more clarification.

There was a discussion about whether flibe is safety-related. I just wanted to point to Table 3.6-1 where the reactor coolant is identified as safety-related.

CHAIR PETTI: Okay.

MEMBER BALLINGER: We have, effectively, done Chapter 3 now, right?

CHAIR PETTI: No, there's Sections 3.2, 3.3, 3.4, and 3.5 coming.

MEMBER BALLINGER: Gee.

CHAIR PETTI: We'll be back at 45 minutes after the hour. Thank you.

(Whereupon, the above-entitled matter went off the record at 10:26 a.m. and resumed at 10:45 a.m.)

CHAIR PETTI: Okay, we're back.

We're going to hear about Chapter 2 and some sections on Chapter 3. Kairos, please start.

MR. BRYAN: Hi, Dave. This is Marty Bryan. I'm the Senior Manager for Site Licensing for Kairos, and I'll be taking us through Sections 2.1 through 2.3. And then, Brian Song, our Senior Manager with Civil Structures, will be covering the remainder of the presentation.

Next slide, please. So, as has been mentioned, the Hermes site is located in Oak Ridge, Tennessee. It's in Roane County and within the East Tennessee Technology Park. And the Hermes test reactor will be located on the former DOE K-33 building site.

So, if you look at the bottom lefthand picture, we're talking about the area in the yellow circle. And just to get you oriented, the Oak Ridge National Laboratory there is off to the right. The actual city of Oak Ridge would be up to the sort of upper right there as well. And down at the bottom, we did reference some material from the Clinch River Nuclear Project, and the Clinch River site is about 3 and a half miles away.

So, if you look at --

MEMBER HALNON: Hold on. The Clinch River site, is that where they did their characterization for the SMRs that they're right now thinking about?

MR. BRYAN: Yes, that's correct.

MEMBER HALNON: Okay. I might have questions later on, but that's fine. I just wanted to get a spatial. Thanks.

MR. BRYAN: Yes, we used some regional geology from that effort and it's about 3 and a half miles away from the Hermes site.

Over on the right hand side is the insert there. You can see the yellow line is the boundary of our property, and you can see that the rectangular field there is the former K-33 facility, at which Hermes will be located down in the southeastern corner. And at the bottom is the former K-31 facility.

And so, it would be, as I said, approximately 185 acres. About 30 acres of this would be permanently disturbed.

MEMBER HALNON: Is there any residual radioactive material there right now from the previous --

MR. BRYAN: Not really. I mean, it's been released for industrial use by the DOE.

MEMBER HALNON: Okay. So, from your perspective on decommissioning down the road, there's nothing to be concerned about? I guess that was a question rather than a statement.

MR. BRYAN: Not from what's there. Obviously, during decommissioning, we would have to address anything from the Hermes site itself.

MEMBER HALNON: Right. Okay. But when you have decommissioning, you have to, I mean, if you're going to release it -- maybe that's not the plan -- of course, you'll have to work through that.

But you have to go below the level of the foundations, and whatnot, to a certain depth.

And this is another question, I guess. And you're confident that there's nothing from previous facilities there that would be of concern when you get to the decommissioning of Clinch -- I mean of Hermes?

MR. BRYAN: That's correct, based on the information we have from the Department of Energy.

MEMBER HALNON: Okay.

MR. BRYAN: And it's been released for industrial use now.

MEMBER HALNON: All right. Thanks.

MR. BRYAN: Next slide, please.

So, I like this slide because it really gives you a picture of what used to be there versus what is there today. So, on the lefthand side, you can see the K-33 building. This was about a 40-acre field. So, a large structure. You can see K-31 off to the right. And then, across Poplar Creek is the K-25, which is now a National Historic Park.

And the K-33 building was originally constructed in 1954. The enrichment facility ceased operations in 1985. DOE began re-industrialization of the whole East Tennessee Technology Park in 1996, and then, it's released for industrial use in 2011.

If you look at the picture on the right side, this is what it looks like today. It's all been reclaimed and it is a true brownfield. This rectangle is the area where Hermes will be, particularly down here in the lower southeastern corner, which you can see in another slide.

Next slide, please. So if you look over at the right hand side, you can see the square down there in the southeastern corner of the former K-33. That's where the Hermes site will be located.

So, just to cover some of the boundary and zone area details, the site boundary is defined by the area owned, leased, or controlled, and the exclusion area is defined as the area within the site boundary where the reactor site management has direct authority over all activities. Our site boundary and exclusionary boundary are coincident, as is the emergency planning zone. They're all coincident to the site boundary.

The low population zone is conservatively set at 800 meters from the reactor. You could see, up in the northwest corner, the nearest resident is approximately .7 miles from the boundary. And then, the PSAR includes population data five miles from the reactor.

Next slide, please. Okay. So we looked at the nearby industrial, transportation, and military facilities within five miles of the site to identify potential external hazards, such as explosions, flammable vapor clouds, toxic chemicals, and fires. And the potential external hazards within five miles were determined to not work for the analysis, with the exception of ammonia and chlorine.

And that's because the distance to Tennessee Highway 58 was less than the safe distance calculated for shipments. Therefore, the main control room will be designated and designed with detectors for these chemicals.

Regarding airports, there's no existing commercial airports within 10 miles of the site. However, a general aviation airport is proposed to be located approximately one mile southeast of the site.

So, we did a screening analysis for the annual probability of aircraft crashing into the facility using DOE Standard 3014-2006. And the total crash frequency for small, non-military aircraft from general aviation or helicopter operations was above the screening, except its frequency threshold. Therefore, the safety-related portion of the reactor building structure will be designed to withstand the impact of small, non-military general aviation aircraft.

Next slide, please.

MEMBER HALNON: So the size of the aircraft is important. Is there enough information on the proposed airport to be able to size that aircraft that you would be using in your analysis?

MR. BRYAN: Yes, good question. There was an environmental assessment performed and it listed the types of aircraft anticipated for the airport. And so, we factored that into our screening analysis.

MEMBER HALNON: Okay.

MR. BRYAN: It had quite a few details about what the proposed aircraft would be.

MEMBER HALNON: Okay. It would be interesting to see what kind of margin you build into that, just in case the airport expands beyond its dream right now.

MR. BRYAN: Yes. There's no construction been started yet, but, yes, you're right, it's been postulated for some time. But if we become aware of additional details, we'll certainly factor that into the operating license application.

MEMBER HALNON: Got it. Thank you.

MR. BRYAN: Okay. Next slide, please.

Okay. This slide is showing -- you know, we're going to talk about Chapter 2 and we mixed in Chapter 3. And the reason we did that is we wanted to show the relationship between the two.

So, in Chapter 2, we're discussing the design basis parameter input envelope. In Chapter 3, we define the methods to translate those inputs into design loads.

And then, in Section 3.5, we define the protections for safety-related SSCs using design loads.

Next slide, please. So the first one of these we're going to talk about is meteorology. As is mentioned -- sorry, was there a question?

CHAIR PETTI: No, keep going.

MR. BRYAN: Okay. So, as we mentioned, the Hermes site is located on the prior Department of Energy Nuclear Facility site within the DOE-managed Oak Ridge Reservation. And the ORR has an extensive network of meteorological towers.

And there was two studies done in 1953 and 2011, meteorological studies. We used newer data. But these studies indicate the basic flow patterns that have been in place during the recorded weather history of the ORR area.

And if you've been to the area, you know that it's influenced a lot by the mountainous terrain, and specifically, the Cumberland Mountains to the northwest and the Great Smoky Mountains to the southeast. And therefore, the prevailing winds in the region reflect channeling of airflow from southwest to northeast, caused by the orientation of the valleys and ridges.

Next slide, please. So extreme winds are based on the climatological data from Oak Ridge and Knoxville and observations from the Met Tower J and L on the ORR. And for a 100-year return period, the maximum wind speed is 90 miles an hour.

Now, hurricane winds are factored in. They're mainly a concern for coastal locations, but we used the contours in Reg Guide 1.221. And the probability of a tornado occurring at the site is low, based on records from the National Weather Service Morristown Tornado Database.

Regarding precipitation, historical precipitation data for the site was obtained from several National Weather Service and Tennessee Valley Authorities sites. Storms with ice greater than 1 inch occurred five times in 50 years and storms with ice greater than 2 inches occurred two times in 50 years. And the maximum historical snowfall event for a 40-hour period was determined to be 28 inches in Westbourne, Tennessee, which is up on the border with Kentucky. And this was back in February of 1916.

Next slide. So if there's no questions, I will turn it over to Brian Song, who will go through the remainder of the presentation.

MR. SONG: All right. Thank you, Marty.

This is Brian Song from Kairos Power. I'm Senior Manager in Civil Structures, and I will present Chapter 2.4, 3.2, and 3.3.

For Chapter 2.4, "Hydrology," the site is located near Clinch River and Poplar Creek, as you see on the right side of the graph. And TVA manages the water levels year around for dam safety and flood control to these two rivers. Both Clinch River and Poplar Creek are considered to be potential flooding resources to our site.

The current grade level that we have is 765 feet above mean sea level. And the normal water surface level at the Poplar Creek, which is on the right side, is about 21 feet lower than our site grade.

Next slide, please. There's two previous flood studies that we looked into. And there's one that's the FEMA Flood Insurance Study for Roane County, Tennessee. This includes 10-, 50-, 100-, and 500-year return periods. All the flood level elevations from these studies indicate that the Hermes site, the flood level is below the Hermes site grade, which is 765 feet above mean sea level.

Another study that we looked into is the Flood Hazard Evaluation for UCOR that they updated on April of 2015. There's a large range of return period that was investigated in this study, from 25-year to 100,000-year, that was modeled and estimated. The results are assessed during our review, and we are identifying the preliminary design basis flood based on these studies.

Next slide, please. So for the credible hydrology event that we defined as a design basis for the Hermes site, it's selected as 25,000-year return period, This is consistent with the Flood Design Category 4, FDC 4, which is a DOE-based criteria.

The results in a design basis flooding level for the site with using a 25,000-year return period result comes up to 759.9 feet mean sea level. This is based on these studies. And that is actually 5.1 feet below the plant grade level of 765 mean sea level.

And also, this Hermes site layout, we have some advantages, that the existing site topography shows the stormwater runoff naturally drains to the east, south, and west of the -- west to flow towards the Poplar Creek.

Next slide, please. So this is about Chapter 3.2 in regards to meteorological damage. The design basis of SSCs is considering meteorological damage, which includes rain, snow, wind, tornado, and tornado- and wind-borne missiles at the site.

And the safety-related portion of the reactor building provides protection to the safety-related systems and components from this meteorological damage. Currently, no credit is taken, based on the PSAR, no credit is taken from the non-safety-related portion of the reactor building, which is the exterior shell.

The design basis meteorological permit applicable to the design of the safety-related portion of the reactor building are established for normal winds and high wind modes, which includes tornado and hurricanes, and precipitation modes.

For the normal wind load design basis, we are following the local protocol which sites ASCE 7-

10, and we are using Risk Category 4, which is for the hazardous substances; that includes in the building. And this is the most stringent design basis that is used in ASCE 7. So, that is used for our safety-related portion of the reactor building.

That results in design basis wind load velocity of 120 miles per hour, and this is based on a 1700-year mean recurrence interval, which is more conservative than the 100-year return period.

Also, we applied normal wind modes that are determined using ASCE 7-10, Risk Category 4, and also, Exposure Category C.

Okay. Next slide, please. And high wind and load are based on two Reg Guides. The first Reg Guide is 1.76, Revision 1. This determines the characteristics of the design basis tornado. And the tornado winds are determined using the same methods in ASCE 7-10 with the wind speeds from Reg Guide 1.76, Revision 1.

And the loads from tornado-generated missile impacts are transformed into an effective or equivalent static load consistent with NUREG-0800, Section 3.5.3, Subsection (ii), using missile spectrum and maximum horizontal speeds provided in Table 2 of Reg Guide 1.76, Revision 1.

The other guidance is Reg Guide 1.221, Rev. 0, and this design gives us the design parameters for hurricane loads. And applied hurricane wind loads are determined also using ASCE 7-10, the maximum wind speed of 138 miles per hour, and velocity pressure is based on the Reg Guide 1.221 for the site location.

Loads from hurricane-generated missile impacts are transformed into an effective or equivalent static load consistent with NUREG-0800, Section 3.5.3, Subsection (ii), using the missile spectrum from Reg Guide 1.221.

All right. Next slide, please. A precipitation note. The grading and drainage design for the site will include loads from precipitation accumulation on the ground affecting the safety-related portion of the reactor building. And the non-safety-related exterior shell of the reactor building has a slope of the roof. So, the loads due to rain accumulation are not considered as a structural load in the structural design.

And similarly, because of the lack of rain accumulation, the load due to ice is anticipated to be very minimal, and is, therefore, enveloped by the snow load that we're going to use. And the snow load parameters are based on Chapter 1 and ASCE 7-10, Risk

Category 4, structures at the site location. And the snow loads that are determined are based on the ground snow load of 21.9 PSF, and we're using the same criteria, based on ASCE 7-10, Risk Category 4, for these structures.

MEMBER BALLINGER: This is Ron Ballinger.

I had a question about the ice load not having to be considered because the roof is sloped. I can tell you from personal experience that in certain temperatures, when it rains, it freezes on the roof; the ice stays on the roof.

MR. SONG: Yes, so we are following the local, the Code of 7-10, and the criteria based on that will be followed for the ice. It indicates it's going to be minimal. So, yes, there will be ice load, but it will be minimal, based on the Code.

MEMBER BALLINGER: My house is not designed to the Code.

MEMBER HALNON: It's also in Massachusetts.

(Laughter.)

DR. BLEY: I think that's the big thing, Ron. I've lived in Upstate New York for a while, and we had to have heating elements on the roof to prevent that sort of thing, but maybe not down here.

MEMBER BALLINGER: Yes.

DR. BLEY: It's Dennis Bley. I have a question that I've never quite understood.

When you look at snow loads, I think you always assume you've got a mass of snow and that's the final load. But almost anywhere, including that part of the country -- and I've lived there -- after you get a big snow, it's not uncommon, eventually, to have some light rain come. And I tell you, that snow gets a heck of a lot heavier. Do you look at that, the load of rain following the snow?

MR. SONG: Yes, it is considered in the Code, as mentioned below, as well. So, unbalanced snow load, snow drifts, and rain on snow, surcharge loads, are also considered, based on the Code of ASCE 7. So, we will be considering that, yes.

DR. BLEY: Okay. I'll have to go look at that, because I in the past have gotten a different answer on this from other people. But, based on what you're doing, it kind of sounds reasonable.

A couple of slides back when you were talking about rain, flooding, you looked at periods like 1700 years and 25,000 years, or something like that. And clearly, there's no collected data going back to those timeframes. Did you use some form of paleo data to estimate if these things have happened in the distant past? How did you come up with those types of flooding numbers?

MR. SONG: Okay. So, I think you're mentioning the slide 2.4, if that is correct --

DR. BLEY: Yes, the slide number is not important, but --

MR. SONG: Got it.

DR. BLEY: Yes. There was another one after this that had a smaller number for a different kind of hydrological event.

But, anyway, yes. So, how do you come up with those?

MR. SONG: I would like to defer that question to Lori, our consultant.

DR. GROSS: Okay. Hi. I'm Lori Gross.

So, it's a complicated, sort of bigger technical response, but, in essence, we do have a historical record of floods and paleo records of floods, but, certainly, they aren't that high. The maximum floods that have been observed are on the order of, I'll say, 70- to 100-year return period floods.

So, what is done is a statistical -- you know, the historical data is evaluated statistically.

There's also some hydrologic models that are used to project up into these lower-probability, higher-return-period floods. So, they are estimated through modeling. Those models are calibrated, I'll say -- I'll use that word "calibrated" -- using historical records.

DR. BLEY: Well, I don't see how you can do this in convincing way unless you look at some kinds of paleo information.

DR. GROSS: It is. It is. There's a lot of information, historically, that's been collected.

DR. BLEY: Okay.

DR. GROSS: And I think that was presented in some previous slides about all the data within the Oak Ridge area. TVA has done a lot of data collection modeling for their designs and, you know, all the work that they do.

DR. BLEY: Okay. As long as you did that, I'm much more comfortable with it. The idea of statistically extrapolating 100 years' worth of data doesn't make any sense to me, but looking at indications in the earth and the surroundings of what's happened in the past certainly does give a basis for it.

But thank you.

DR. GROSS: Sure.

MEMBER HALNON: This is Greg Halnon.

One more piece of operating experience on your flooding aspect and the ice is ice on the roof, followed by rain, melts faster than ice on yard drains that sit in the shade. So, if you're going to credit any yard drains to help get that precipitation off, keep the timing of that in mind relative to a freeze/thaw event.

DR. SCHULTZ: This is Steve Schultz.

Just to comment, to follow up on Dennis' note, where you well could have snow following by rain. Given that you're using the maximum event associated with snow, 28 inches, and then, have rain on top of that, the 22 pounds per square foot really does seem low in that calculation. It should be a higher value for that.

MR. SONG: Yes. So, the value of the ground snow is there, but, yes, based on the Code, we will evaluate the correct value.

DR. SCHULTZ: Good. Thank you.

MEMBER KIRCHNER: Brian or Lori, following up on Dennis' question, do you have to look at upstream dam failures and the TVA watershed there?

DR. GROSS: That is a requirement.

Brian, I'm assuming you want me to answer that?

MR. SONG: Yes. Yes, please do.

DR. GROSS: Okay. Sorry.

Yes, this is Lori again.

That is a requirement and that was presented in the -- there was some of that work done in those historical studies that Brian cited that are in the PSAR. And I'm saying, yes, that is considered. It was not factored into the design basis flood, but it is part of the evaluation.

MEMBER KIRCHNER: I think, if I remember correctly, when we reviewed the Clinch River early site permit application, I think that was also considered in their assessment for their site, which isn't too far away from you.

DR. GROSS: Yes.

MEMBER KIRCHNER: Okay. Thank you.

DR. GROSS: Okay. Is that sufficient then?

Yes, I agree with you that is a big -- that is a standard requirement, to look at flooding hazards due to precipitation and, separately, hypothetical, I'll say, simulated dam failures.

MEMBER DIMITRIJEVIC: This is Vesna

Dimitrijevic.

I just have one comment just to let you know that -- are you saying that this will be addressed in the OL, the rain on the snow and snow movement after construction is finished and the slopes of the roofs?

I just want to make something for good keeping. They didn't make it to say it is one of the OL expectations. So, it's critically missing. So, it would be nice to audit. Okay?

MR. SONG: Sounds good.

All right. So, I'll continue.

Next slide, please. All right. So, the safety-related portion of the reactor building considers load from both external and internal flooding events.

For the external flooding event, there's no pose of hydraulic load because the grade elevation is above the design basis flood elevation determined in PSAR Section 2.4.

And the internal postulated events consider the water resources within the safety-related portion of the reactor building.

So, it will be discussed in Section 3.5, but the safety-related SSCs are protected from internal flooding. For instance, the safety-related SSC that are vulnerable to flooding are elevated or shielded, or otherwise protected from spray. And this includes the flibe-bearing components.

Design features direct water flow and prevent it from entering enclosures containing safety-related SSCs. And the volume of the water in the safety-related portion of the reactor building is limited by the design. And the water systems that cross the base isolation mode automatic or manual termination of flow will be specified in the operating license application.

So, I think that ends my slides.

CHAIR PETTI: Okay. Oh, go ahead.

MEMBER BIER: Yes, I have a question going back to Chapter 2. I don't think we need to put the slides back up.

But, in Chairman Rempe's absence, she encouraged me to take a look at the population projections. And it looks like, based on demographic data for the counties, you folks are predicting a decline in population, which seems to be accurate. The 2020 Census has a smaller population than 2010, and, you know, not unexpected in a sort of rural area.

But, within the five-mile radius, there

are a number of new facilities going in -- airport, Clinch River pharmaceutical company, et cetera. So, the daytime population in those areas may actually be going up quite a bit.

And I understand the reasons for wanting to rely on kind of official published demographic data, rather than doing your own estimates, but, in this case, it may not be too realistic to do that, you know, within the five-mile boundary. So, I just wanted to raise that as a household issue.

MR. BRYAN: Yes, this is Marty Bryan.

Yes, that's a good point. We will look at that in the operating license application to see if there's an update.

MEMBER BIER: Okay. Thank you.

CHAIR PETTI: Okay. I don't see any other hands up.

So, staff.

MR. HELVENSTON: Our presenters for this section are virtual. So, I believe you're going, first, Amit. So, I'll turn it over to you.

MR. GHOSH: Yes. Thanks, Ed.

Good morning, everybody.

My name is Amit Ghosh, and I will start the presentation of the staff review of PSAR Sections

2.1 through 2.4 and 3.2 and 3.3.

I am from the Division of Engineering and External Hazards of the Office of Nuclear Reactor Regulation.

Next slide, please. I will be presenting, first, to the 2.1, "Geography and Demography," and 2.2, "Nearby Industrial Transportation and Military Facilities." And I will come back at the end to present 3.2 and 3.3 on behalf of my colleague who did the actual review, Bryce Lehman. I'll just present it because, unfortunately, he is not able to make it in the morning. He'll be there in the afternoon.

Next slide, please. So, first, I have listed the regulatory vessels which we have used in this review, 50.34(a), 50.35, 50.40, and 100.10, and we used the Standard Review Plan and NUREG-1537.

For Section 2.1 only, we also used 10 CFR 100.11(a) for determination of exclusion area, low population zone, and distance to nearby population centers. We also used Reg Guide 2.6, Rev. 2, and ANSI/ANS 15.16, "Emergency Planning for Research Reactors."

For Section 2.2, we used Reg Guide 1.21 to evaluate the determination of explosions postulated to occur at nearby facilities or a transportation route.

Next slide, please. So you have in the morning, a few minutes back, Kairos' presentation, very nice pictures of the site location. It is the old K-31 and K-33 facility of the Oak Ridge Reservation Gas Diffusion Plant which was used to enrich uranium at one time.

The site near the reactor is relatively flat. We reviewed the exclusion area boundary, low population zone, emergency planning zone, and we found they are reasonably done.

We also looked into the nearby population center. And based on 2020 Census, the city of Oak Ridge is the one within -- because the definition of population center more than 25,000 residents. So, the city of Oak Ridge has 31,402 residents at Census. And it turns out this site also is within the city limits, but, as we discussed, the nearby resident is about .7 miles away.

There is very low population close to the facility. Most of the people live at the northeast corner of the map which was shown before. So, we measured the distance and we found the site also passes the NRC criterion, 1 and one-third of the EPZ for the nearest population center.

And we checked the population projection which Kairos used from the Boyd Center for Business and Economic Research of Tennessee, which is a reliable organization.

Next slide, please. Is there any question on 2.1? Because that's all I have to present. Otherwise, I will go to 2.2.

CHAIR PETTI: Keep going.

MR. GHOSH: Okay. Kairos has used the Site Safety Analysis Report, SSAR, of the Clinch River Nuclear Site ESP application. They used their analysis because very similar things are here.

And as Marty discussed a few minutes back, there is a new airport proposed to be constructed very close to this site. That's called -- the name is Oak Ridge Airport. It is supposed to be -- construction is supposed to start next year, 2024.

And if you look into this map, you can see the Clinch River and the Kairos Hermes site. There is a highway, Tennessee 58. The runway will be parallel to that. So, it is about less than a mile from the reactor site.

And what we understand is that this new airport, it will be a general aviation airport, mostly for the business jets, people coming to Oak Ridge, and doing their business, and going back. Ninety-seven percent of the flights are estimated to be general aviation aircraft. Only about 3 percent will be helicopters, which may be local, and very few, and used for spraying or medical purposes. So, for design purposes, this general aviation airport will be a light one. And I'll talk a little bit more about that in the next slide.

Next slide, please. So Kairos has looked into the possible or potential detonation of high explosives, the explosion of chemical vapor clouds, release of toxic chemicals, and fires while being transported nearby to highways, railways, waterways, or carried in pipelines, or when they are stored at the nearby facilities.

They assessed the potential hazard area and checked each of them, whether the distance is enough not to be a credible hazard; whether the quantity of hazardous material released is so small that it doesn't a big difference, any much of a difference, or the potential consequences are tolerable.

Based on their analysis -- and we have gone through those analyses and agreed that -- the potential release of chlorine and anhydrous ammonia, when being transported very close to this Highway 58, Tennessee 58, could be a problem to the main control room. So, they will be equipping the ventilation system at the main control room with chlorine and ammonia detectors.

We also looked into -- I mean they also looked into the potential annual frequency of aircraft crashes while landing or taking off from this future Oak Ridge Airport. We checked the FAA database and they are listing it as a new airport coming in soon.

The FAA -- or DOE environmental assessment, as Marty has talked a few minutes back, and they have identified which craft, 500 aircraft is the typical one which they considered in designing the runway, and, also, my guess is that would be a very nice and appropriate aircraft to use in our design process of the structure.

As Kairos has already said, the site doesn't pass the initial screening criterion. So, they will be designing the safety-related portion of the reactor building to withstand a crash. So, in the OL application, we'll be looking at the loads taken and other details.

So, these are my findings and that is also the regulatory findings, that we found the information present in PSAR Section 2.2.1 and 2.2 is sufficient and meets the applicable guidance and regulatory requirements to issue a construction permit, in accordance with 10 CFR 50.35.

MEMBER HALNON: Amit, this is Greg Halnon.

In the aircraft crash into the reactor building, if that's going to be in the design basis, the ensuing fire is still, in my mind, part of that. And previously, we didn't ask -- we didn't require that requirement because 10 CFR 50.155, or something up in that range, was not required for that facility.

As you get to the operating license, I would encourage you to ask the question, or at least internally, whether or not the ensuing fire is a concern and to make sure that that site will be able to withstand that.

MR. GHOSH: Yes. And I give a very good example of 9/11, the twin towers. Yes, they got hit, physically damaged by the crashing aircraft, but the whole structure collapsed because of the actual fire, the heat, the thermal part of it, based on the analysis done by NIST after that.

So, when the time comes, we'll be asking all those questions and look at it. And I have to, obviously, rely on my structure colleagues, thermal areas, on that one.

MEMBER HALNON: And the fire protection.

MR. GHOSH: Yes, fire protection.

MEMBER HALNON: Thank you.

MR. GHOSH: Thank you.

MEMBER BIER: I have another question. This is Vicki Bier.

With regard to the chlorine and ammonia detectors -- again, this is kind of more of a question probably for the operating license phase -- but is it anticipated that control room operators could evacuate and leave the facility in a safe condition? Or do we need a system that can somehow isolate the control room for a period of time, so they can remain there? Do you know what's anticipated for operation under those conditions?

MR. GHOSH: Yes, very good question. At this moment, with the kind of details we have, I cannot answer this question. All we know, like this facility, the main control room may be affected if there is a chlorine and ammonia spill very closeby. So, once the time comes, we are going to look into that, the design of it and how -- all the details.

MEMBER BIER: Sure. Is that something that Kairos wants to comment on at this moment?

MR. PEEBLES: We don't have any comment at this time.

MEMBER BIER: Yes.

MR. PEEBLES: We'll provide more details at the OLA phase.

MEMBER BIER: Okay. Thank you.

MR. PEEBLES: Sorry. That was Drew Peebles, Senior Licensing Manager.

MR. GHOSH: Is there any other question for me?

Then, I will hand it over to the next presenter on Section 2.3, "Meteorology."

MR. WHITE: Thanks, Amit. Good morning. My name is Jason White. I'm a Meteorologist in the External Hazards Branch here at the NRC.

Today, I will be discussing Section 2.3 of the application which focuses on the site characteristics related to meteorology.

The regulatory basis for the review of this section was mentioned previously in the presentation. In addition, the staff used the guidance of NUREG-1537, Part 2, Section 2.3, which is the section focusing on meteorology.

The Applicant provided information related to the original climatology and local meteorology of the proposed site. Information in these sections included the site characteristics related to wind, temperature, precipitation, atmosphere stability, and various extreme weather conditions.

The NRC staff reviewed a description of the general climate of the region and the meteorological conditions relevant to the design and operation of the facility. This included the staff reviewing the various data resources and the analytical approaches used by the Applicant to prepare the information.

Based on this review, the NRC staff concludes that the site characteristics associated with the regional climatology and local meteorology are representative of the region at the proposed site, and therefore, acceptable for use in evaluating the conditions at the site.

Next slide.

MEMBER HALNON: Hey, Jason, this is Greg Halnon.

Did you, in your review, compare what the ESP was with Clinch River, since it's so close? I mean, did they use all the appropriate HMRs and whatever analytical models were out there to make sure they're at least within the ballpark of the facilities there nearby?

MR. WHITE: What was the first part of your question? I missed it. I'm sorry.

MEMBER HALNON: I mean, clearly, there had to be some met data from the ESP at Clinch River, which isn't too far away. And I realize that, from a flooding perspective, it's closer to a river, and whatnot. But, for the intense, you know, the precipitation models and storms and transpositions, and all that stuff, was that compared to what the Clinch River ESP had done?

MR. WHITE: I didn't necessarily compare it to the Clinch River data, but we did look at the representative data from the National Weather Service and other local data sources within the region. I'm sure it's probably similar to the data that they actually pulled from Clinch River.

MEMBER HALNON: Okay. So, if we compare the two, we shouldn't see anything that's way out different from a precipitation, intense precipitation, or storm transposition aspect?

MR. WHITE: Correct. I think that it should be similar. I'm not sure exactly in terms of the distance between the two, if there are any topographical effects that would change what we would expect to see, but, overall, for the site, since they're in the same general area, I would expect to see similar results.

MEMBER HALNON: You know, I think it was only three-plus miles, 3 and a half miles away. So, okay. I'll take a look. It's more of a curiosity.

MR. WHITE: Okay.

MEMBER HALNON: Because there would be a lot of work that had already been done on the ESP that should be applicable here. Thanks.

MR. WHITE: Yes. So, on this slide, the Applicant also provided information related to the meteorological monitoring program and atmospheric dispersion modeling at the proposed site.

Information in these sections included a description of the location and measurements taken at the network of meteorological towers, as well as the methodology used for the atmospheric dispersion modeling.

The NRC staff reviewed the information describing the network of meteorological towers and their measurement capabilities. The staff reviewed the data taken from the towers to support the dispersion analysis at the proposed site and the methodology, inputs, assumptions used in the short-term atmospheric dispersion analysis.

For the long-term atmospheric dispersion analysis, Kairos states in the PSAR that modeling for the routine releases will be provided in the operating license application.

Based on this review and its confirmatory analysis of the atmospheric dispersion factors, the NRC staff concludes that Kairos' meteorological monitoring program is acceptable; that the meteorological dataset provided is representative of the proposed site, and sufficient to support dispersion analysis, and that Kairos' short-term atmospheric dispersion analysis and calculated dispersion factors for the EAB and outer boundary of the LPZ are also acceptable.

Next slide. The NRC staff concludes, based on a review of the information provided in the application, subsequent RAI responses and the staff's technical evaluation, that the information related to meteorology in the Hermes PSAR Section 2.3 is sufficient and meets the applicable guidance and regulatory requirements for the issuance of a construction permit, in accordance with 10 CFR 50.35.

Further information on meteorology, namely, the details regarding the long-term dispersion modeling for routine releases, can reasonably be left for later consideration in the operating license application.

So, that concludes the presentation on meteorology. Are there any further questions before I turn it over to Yuan?

CHAIR PETTI: Keep going.

MR. CHENG: This is Yuan Cheng. I'm a hydrologist of the NRC. I am a hydrologist and responsible to evaluate the Hermes site safety evaluations related to the PSAR Section 2.4, "Hydrology."

The Applicant provided information on the following. The staff evaluated the hydrologic descriptions and historical flood records. The staff finds the design basis floods is the result of the flood hazards analysis, based on the industry-accepted methods.

The staff evaluated hypothetical dam failures in the upstreams. The staff finds such an event is highly unlikely for the four years operational life of the Hermes reactor. This finding is based on TVA providing the dam inspections and performing dam maintenance for TVA dams.

The staff evaluated the groundwater and finds the groundwater is protected, based on the following: the groundwater levels, in general, it's about 10 feet below the grade of the Hermes site. There is no groundwater extraction and injection to operate the Hermes reactor.

The (audio interference) of the reactor coolant is confined in the plant building. The fission products, such as tritium, are monitored and confined in the plant.

And environmental monitoring program will be implemented to detect any radiological releases beyond normal operational release.

Next slide, please. The staff finds the Hermes site elevations provides approximately 5-feet safety margins above the design basis for elevation for the stream and river floods. The design basis flood elevation for the stream and river floods is higher than FEMA's 5 feet of flood elevation by approximately 10 feet.

The site presents no significant risk to the Hermes facility due to the postulated flood events.

The facility's design basis includes mitigation and prevention of uncontrolled leakage or loss of the reactor coolant to groundwater and Service water.

Based on the above findings, the site hydrology demonstrates that the Hermes site is an adequate site to support facility design basis and satisfies the applicable acceptance criterias of NUREG-1537, Part 2, Section 2.4.

Next slide, please.

MEMBER HALNON: Yuan, this is Greg Halnon again. Are all those conclusions going to be confirmed with site characterization? Or has that already been done?

MR. CHENG: It's already done.

MEMBER HALNON: Okay. So, we know for a fact that the 10 feet --

MR. CHENG: Yes.

MEMBER HALNON: -- below the grade --

MR. CHENG: Yes.

MEMBER HALNON: And when you say, "below the grade," is that below the lowest point of the buildings from a (audio interference) perspective or --

MR. CHENG: The site is flat. I believe that the floor elevation maybe has a couple of inches above the current grade, because they will have constructions, like a concrete slab. In general, in the construction, they may have 9 or 6 -- 6 inches to

9 inches is the floor thickness. So, they will be above the grade by that, the levels. So, in general, the 10 feet above the 500-year flood is correct.

MEMBER HALNON: That's a generalization --

MR. CHENG: Yes. Yes, it may be plus or minus .5 feet.

MEMBER HALNON: Okay. Thank you.

MR. CHENG: The staff findings, there is reasonable assurance that the preliminary information conforms with 10 CFR 110 and supports 10 CFR 50.34 by providing flood hazards analysis and site evaluation factors.

The staff concludes the information of hydrology in Hermes PSAR Section 2.4 is sufficient and meets the applicable guidance and the regulatory requirements identified in these sections for the issuance of a construction permit, in accordance with 10 CFR 50.35. And additional information on hydrology can reasonably be left for later consideration in the operating license application.

This is the end of my presentation. Do you have any other questions?

I will pass the next presentation to my colleague, Amit.

MR. GHOSH: Hi. Thank you, Yuan

This is Amit Ghosh again. I'll be talking on behalf of the reviewer, Bryce Lehman, on PSAR Sections 3.2 and 3.3, "Meteorological and Water Damage Review."

As we heard a few minutes back, all the safety-related structures are designed to withstand meteorological loads and the water loads, including internal and external floods.

Next slide, please. So, there are three regulatory bases which we used in this review: 50.34(a), 50.35, 50.40, and there is an approved design Topical Report, Technical Report, Primary Design Criteria 3 -- sorry -- 2, Design Vessels for Protection against Natural Phenomena in the Approved Report KP-TR-003-NP, the (audio interference) version.

And we used the relevant guide NUREG-1537, Section 2.2, "Meteorological Damage," and Section 2.3, "Water Damage."

Next slide, please. We have verified that the structural loads are approved for rain, snow, and wind using the ASCE guidance 7-10 with the Risk Category Structure 4, the highest risk category. The tornado wind loads and the associated missiles come from our Reg Guide 1.76 for Region I. For hurricane load, the Reg Guide 1.221 gives the appropriate wind speed and, also, the missile characteristics. So, those have been taken into account in designing that.

And we also find in our review what are the flooding-related loads. They are appropriately propagated from Chapter 2 review. The reactor building, the safety-related portions were designed in accordance with National Standard or American Concrete Institute 349 and American Institute of Steel Construction ANSI/AISC N690 -- very, very widely used that. And we also considered it to meet the Primary Design Criteria 2.

Next slide, please. As Yuan and Kairos just presented, the extended flood level will be below the grade of this structure. All the SSCs important to safety will not be below that. And there will be grading which will try to take the water away from the reactor, which, generally, everybody uses -- like don't want the water to be near the basement.

So, we'll be reviewing the site features and the design, how that thing will be carried out. And there will be limited water available to minimize the potential for internal flooding. The SSCs in the building will be (audio interference). There is not much of flooding issues. And there will be, obviously, drains to minimize that.

We will be reviewing the specific details of the layout and the amount of water that may have the potential to be released during the OL review phase.

Next slide, please. So in conclusion, the safety-related portions of the reactor building are designed for good, appropriate codes and standards. Meteorological data has been appropriately provided from Chapter 2 to design this facility, and the meteorological and the hydrological, both water and the weather-related.

The specific design details will be reviewed during the OL phase.

Next slide, please. So we found, with reasonable assurance, that the preliminary information provided in the PSAR, these two sections, conforms with 10 CFR 50.34(a)(iii), by providing information related to the materials of construction, general arrangement, approximate dimension, which is sufficient to provide reasonable assurance that the final design will conform to the design basis.

The staff concludes the information presented in these two sections, 3.2 and 3.3, is sufficient and meets the applicable guidance and regulatory requirements identified in the issuance of the -- ready and can issue the construction permit, in accordance with 10 CFR 50.35, and that information can reasonably be left for the OL application for us to review.

With this, I conclude my presentation on behalf of my colleague. And whatever questions you have, I will try my best to answer them.

CHAIR PETTI: Any questions, Members?

MEMBER KIRCHNER: Amit, this is Walt.

MR. GHOSH: Yes?

MEMBER KIRCHNER: This is more for Kairos. I meant to ask this earlier. I think this is the case, but I'll ask it specifically. Is the decay heat removal system enveloped by the safety-related portion of the reactor building? And it's a leading question because, in other advanced designs, the decay heat removal systems that were also thermal siphon designs had vulnerabilities to external hazards, like wind-driven missiles and aircraft.

MR. PEEBLES: This is Drew Peebles with Kairos.

Yes, the safety-related portions of DHRS are in the safety-related portion of the reactor building.

MEMBER BROWN: What about the water tank?

MEMBER KIRCHNER: Yes, that was the question.

MEMBER BROWN: That feeds the DHRS.

MR. SONG: Yes, that is also included in the safety-related portion.

This is Brian Song.

MEMBER BROWN: Okay. This is Charlie Brown following up with Walt. Sorry. Sorry to interrupt you, Walt.

MEMBER KIRCHNER: Yes. Excellent. Okay. Thank you.

CHAIR PETTI: Okay. Then, hearing no more questions, we're only three minutes behind. It's time for lunch. So, we will again recess, and we will be back at 12:45 Eastern to start with Section 2.5, 3.4, and 3.5. And then, we'll do Chapters 2 and 3 memos, and then, get into Chapter 4. Thank you all.

(Whereupon, the above-entitled matter went off the record at 11:49 a.m. and resumed at 12:49 p.m.)

CHAIR PETTI: Let's get going. Kairos, you're up.

MR. SONG: All right, thank you.

This is Brian Song again, from Kairos Power. I'm a senior manager civil structures. I hope you guys all enjoyed your lunch. Let me continue with Chapter 2.5, 3.4 and 3.5

Next slide, please. So, yeah, as Marty mentioned, Chapter 2 and Chapter 3 are kind of related. And Chapter 2 defines the design basis and parameters.

And Chapter 3 defines the methods of transferring it to the design mode. And that all goes to Chapter -- sorry, Section 3.5 which will define how these goals are being protected, how the SSCs are being protected by, by the design modes.

Next slide, please. So Section 2.5 talks about geology, seismology, and geotechnical engineering. They characterize the geologic, geophysical effects and, again, geotechnical aspects of the region and seis -- to develop a seismic design basis for the facility.

Hermes PSAR relies on existing information from the Clinch River Early Site Permit application for the regional and local geologic description, with supplement information as needed. And that covers 200 miles around the site.

Also, the Clinch River nuclear site is very close to the Hermes site, which is about 3.5 miles. So, we share the same regional geology.

The Hermes probabilistic sites and hazard analysis is also adapted for the -- from the Clinch River Early Site Permit application, PSHA supplemented with consideration of the current site and the publications for the site and regional area are considered.

The PSHA methodology is an enhancement over the guidance of NUREG-1537. And also the Clinch River nuclear site PH -- PSHA meets ANS 2.29 criteria.

Okay, next slide, please. Talking about the site geology, the Clinch River nuclear site's geology information does -- are applicable to our site since it's in close proximity. But a subsurface stratigraphy is -- was also developed for Hermes site from geotechnical boring that we performed.

And the placing at the facility on the site was also informed by the geotechnical information that we gathered.

Next slide, please. For the vibratory ground motion, the Clinch River PSHA, we used it, we used the Clinch River PSHA to develop the site and the design response spectra. And the analysis relies on the information from, also from the Clinch River Early Site Permit, Early Site Permit application, with supplements.

The use of the Clinch River Early Site Permit application on PSHA is both appropriate and reasonable give that we are pretty close to between Clinch River and the Hermes site.

The seismic source transition is based on the Central and Eastern United States seismic source characterization report. And the design response spectra meets ASCE 43.19 and uses Seismic Design Category 3 for the safety-related SSCs, which is appropriate for a non-power reactor application.

Okay. And you can see the graph on the right side. And the performance goal we are aiming is 1E to the minus 4, which is SDC-3.

Next slide, please. For the subsurface deformation, this information also relies, we will also rely on the Clinch River Early Site Permit application, supplemented by the site-specific assessments, with potential -- with a potential of sinkholes, faults, or soil liquefaction.

Given the conditions that we discussed over the foundation interface plans along with fill placement, there is, there is no potential for liquefaction at the site. Only inactive surface faults have been documented within the site area. And the foundation rock for Hermes reactor is at depth at which no evidence of karst dissolution was encountered -- is encountered.

Okay. Next slide, please. For the foundation interface, as you can see on the right side, the foundation layout has been established based on what we know, has been -- based on the knowledge of the site surface conditions gathered from historical documents, and also Clinch River Early Site Permit application, and also the subsurface boring exploration that we did.

The bearing system for the safety-related structure is a foundation mat resting on concrete fill over the bedrock that you see on the right side.

Okay.

MEMBER HALNON: This is Greg.

On the comparison to the Clinch River, is the soil subsurface the same in the -- I thought the Clinch River site was right on the river, and this is quite a ways away. Does that make any difference? Or is that comparable?

MR. SONG: Thanks. Thank you for that question.

I'll rely on one of my consultants. Antonio, if you can.

MR. FERNANDEZ: Sure. Sure.

Hi. This is Antonio Fernandez. And I help Brian with geotechnical and seismic issues.

Yes, that's a good question. Well, in terms of specifically what's beneath the site itself, of course they are different locations and the soil contents are going to be different. So, what we didn't, we didn't rely on the RN on Clinch River for the specific soil beneath Hermes.

So, there's a due diligence investigation and boring investigation to define the soil profile here. So, yeah, in this specific case, this specific picture it's standalone. It's not relying on Clinch River.

MEMBER HALNON: Okay. So, Clinch River informed the --

MR. FERNANDEZ: Right.

Other, other aspects, like, Brian mentioned, like the regional geology, which is kind of a more, much more wider area where there's probably six sites in the analysis, that is, that is reliant on Clinch River. But this is very specific to Hermes.

MEMBER HALNON: Okay, thank you.

MR. FERNANDEZ: Uh-huh.

MR. SONG: All right, thank you.

Any other questions? Or I can proceed to

3.4.

MEMBER HALNON: Go on.

MR. SONG: Okay, right.

So, Chapter 3.4 discusses about the seismic. And we are using the graded performance based approach from ASCE-43.19 in our design to use the design to protection for the safety-related SSCs from the design basis earthquakes.

The safety-related SSCs are designed to SDC-3. Some safety-related SSCs are designed to the local building code, which is consistent with the NUREG-1537 and IAEA TECHDOCS. That is mentioned here.

The return period associated with the design basis ground motion corresponding to SDC-3 is similar to the maximum earthquake specified in building codes with a 2 percent probability of exceedance in 50 years. And this is consistent with the NRC approvals for other non-power reactors.

And it has also additional margin exists due to the short period of operation time that we have for Hermes.

Seismic performance criteria --

CHAIR PETTI: But just a question.

MR. SONG: Yes. Go ahead.

CHAIR PETTI: That 2 percent, that's 4

times to the minus 4. In your earlier sites you had 10 to the minus four as a SDC-3 performance goal. Am I mixing up numbers?

MR. SONG: Correct.

DR. KOSBAB: Brian, would you like me to speak to that?

MR. SONG: Yes.

DR. KOSBAB: Okay. Hi. I'm Ben Kosbab, seismic and structural consultant helping Brian and others at Kairos.

What you saw on the previous slide about 10 to the minus 4 was the structural performance target attached to reliability of the structure and the return period of the earthquake, or the annual exceedance rate to the earthquake.

Here, the 2 percent probability of the exceedance in 50 years is referring to the earthquake return period.

CHAIR PETTI: Okay.

MR. SONG: Okay. All right, so, I'll continue.

The seismic performance criteria are consistent with ANS 15.7, which is reactor, research reactor site evaluation.

The 5 percent damped horizontal and

vertical design response spectra are developed consistent with ANS 2.29, using the design response spectra as defined in Section 2.5.

The structural design of non-safety related SSCs is performed in accordance with the 2012 International Building Code and the Tennessee Building Code.

Okay. Next slide, please. The analysis models, we will be developing a 3-D fine element model for safety-related structures that will be used for seismic analysis. And that is consistent with ASCE 4.16.

Cracking analysis applies based on ASCE 4.16, Table 3-2.

Structural damping applies based on ASCE 4.16, Table 3-1.

The structural mass captures self-weight of structural elements as well as portion of the design live loads and the design of uniform snow load.

The model uses three component seismic inputs to develop the structural forces and end structure of the response spectra. This is used for SDC 3 structural and equipment qualifications as well.

The seismic response analysis meets ASCE 4.16, Chapter 4, using deterministic linear analysis.

The soil structure analysis will be consistent with ASCE 4.16, Chapter 5.

Okay. Next slide, please. For seismic instrumentation, seismic instrumentations will be installed for monitoring seismic events. And triaxial time history accelerometers will be located in the pre-field and in the safety-related portion of the reactor building.

Okay. Next slide, please. So, okay, so to explain a little bit about the reactor building here. Currently in the PSAR it is 200 feet long and 100 feet wide.

The roof is not shown here, but it is, it will be a sloped roof.

And the safety-related portion of the building uses base isolation, which is a spring, spring-backed element type isolators.

And as you see here, the right side is the reactor cell, which includes vessel, applied inventory, and heat reduction regulator, PSAR. And the fuel cell includes the PHSS, which is the pebble handling and storage system. And spent fuel storage will be on the fuel cell side.

And these will be base isolated.

The other buildings, including the building that is, that houses the main control room, are non-safety related, so.

Next slide, please. All right. The safety functions of the safety-related portion of the reactor building does protect, does protection of safety-related SSCs from design base natural phenomena and external hazards.

The structural support for the safety-related SSCs is located on the safety-related portion of the -- are located in the safety-related portion of the reactor building.

Protection of the adverse effects of non-safety related SSCs relies on the ability of safety-related SSCs to perform their safety functions and prevent interactions between reactor coolant and the water contained in concrete in the safety-related portion of the reactor building.

No part of the reactor building is credited to meet the functional containment safety function.

All right. Next slide. It describes the design using prevention standard and in accordance with the applicable quality assurance program based on ASCE 710, SDC 2. Describes protect safety-related SSCs from the effects of design basis meteorology, cold, flooding, and seismic events.

SDC 3 describes design minimize -- design minimizes probability and the effort of fire and explosions. These have low combustible materials operating in the fire protection program.

SDC 75 describes to design protects the geometry of the decay heat removal system from related natural phenomena events. And the DHRS is located in the safety-related portion of the reactor building.

SDC 76 describes design permits, periodic inspections, and survey events of safety-related structure areas. And this is to be determined in the final safety analysis report.

Okay. Next slide, please. All right. Just a little bit of explanation of SDC 2, seismic events. The safety-related portion of the reactor building is reinforced concrete structure, design based on ACI 349-2013. And the internal steel structures are designed to meet AISC N690.

By meeting ASCE 43.19, the safety-related portion of the building provides protection to safety-related SSCs from the design basis earthquakes.

The seismic assessment is checked for both strength and displacement base criteria, and limit states are set based on the target performance goal.

The safety-related portion of the reactor building uses a spring-backed seismic isolation system which lowers seismic demands on the safety-related reactor building and safety-related SSCs in both horizontal and vertical directions.

The moat is sized to accommodate the displacement consistent with the isolation system, meeting the performance goal of 1E minus 4 per year.

And design features a combination, a potential differential displacement for SSCs that cross the moat.

I think that concludes my presentation. If there's any questions?

MEMBER KIRCHNER: This is Walt Kirchner. Just a simple question.

For those components that are non-safety related within the reactor, the safety-related portion of the reactor building, do you use the same analysis tools?

MR. SONG: So, wanted to make sure I understood the question.

Is it do you mean the non-safety related equipment inside the safety-related portion of the reactor building?

MEMBER KIRCHNER: Yes.

MR. SONG: Is that what you are --

MEMBER KIRCHNER: Yeah.

MR. SONG: Yeah.

MEMBER KIRCHNER: It could be just simple yes or no. I, I would expect you would use the same tools.

MR. SONG: Correct, yes. We will use the same tools to --

MEMBER KIRCHNER: Excellent.

Okay, thank you.

MR. SONG: Uh-huh.

CHAIR PETTI: Okay. Seeing no more, let's go to the staff.

MR. HELVENSTON: I believe Jenise Thompson will be starting out virtually for the staff on the next presentation.

MS. THOMPSON: Good afternoon. My name is Jenise Thompson. I'm a geologist in the Division of Engineering and External Hazards in the Office of Nuclear Reactor Regulation.

So, I'll be presenting the staff's review of PSAR Section 2.5, and then handing off this presentation to my colleagues to overview the staff's review of 3.4 and 3.5.

Next slide, please. In Section 2.5 we'll cover the geology, seismology, and geotechnical engineering. And 3.4 we'll address seismic damage. And 3.5 we'll address the reactor building structure.

Next slide, please. The overview of the staff's review is as follows. The applicant applied for a construction permit and has not specifically requested approval for detailed design information.

The applicant also provided a preliminary design description and a discussion of the relevant design bases, for example, the principal design criteria, or SDC.

The staff were asked to assess whether the preliminary design information, including the identification of relevant design bases, was sufficient to allow the staff to determine that the information meets the applicable regulations for the issuance of a construction permit and, also, the detailed design information can be left to the operating license application.

Next slide, please. The regulatory bases for these sections, there's a fair amount of overlap for all sections: 10 CFR 50.34(a), 50.35, and 50.40 apply.

The staff also followed the relevant guidance in NUREG-1537, which is the standard review plan and acceptance criteria for non-power reactors.

For Section 2.5 only, the regulatory requirements in 10 CFR 100.10 are also applicable. And applicable to just Section 3.5 are the principal design criteria that were just discussed of PDCs 1, 2, 3, 75, and 76.

Next slide, please. PSAR Section 2.5.1 and 2.5.2 address the regional and site geology. The regional geology was incorporated by reference from the Clinch River Early Site Permit location, which is 3.5 miles away from the proposed Kairos location.

The Kairos site is underlain by the Mascot dolomite, the Murfreesboro limestone, and the Pond Springs Formation, all of which occur within the Appalachian Valley and Ridge physiographic province.

And as two of the three names suggest, each of these units is to some degree calcareous and, therefore, these foundation rock units are susceptible to karst.

The applicant noted that although there was no surface indication of sinkholes at the site, there was evidence of karstic activity encountered in the borings performed for the location.

Next slide, please. The staff determined that the characterization of the local and regional geology is adequate. We also noted that the potential for surface faulting is something that can be deferred to the operating license.

We also included the geologic mapping permit conditions which, combined with the removal of the overburden soil and weathered rock at the site location, will ensure no evidence of karstic dissolution at the foundation level.

We've included the full text of these geologic mapping permit conditions here, essentially requiring detailed geologic mapping and examination of any encountered geologic features. And then notification to the Director of NRR once those excavations are open for examination by the NRC staff.

And we included the exact permit conditions in the Clinch River ESP as well.

DR. BLEY: I'd like to ask a question. This is Dennis Bley.

I thought when the applicant was talking they said they didn't find any karst. But you said, but they did, you're saying.

Were there indications of, are there big voids underground or is this just small areas?

MS. THOMPSON: In our smaller voids that were encountered in the borings that were performed at the site there was no evidence of large scale sinkhole activity.

But the borings are essentially a thin hole picture of what's going on in the subsurface, which is why we include the geologic mapping permit conditions to ensure that there are no large karst features encountered at the foundation level once that excavation is complete.

DR. BLEY: Makes sense. But these were fairly small it sounds like, and no surface, well, several times no surface vents. Okay, thank you.

MS. THOMPSON: You're welcome.

Next slide, please. Moving into PSAR Section 2.5.3 addressing vibratory ground motion at the site, the applicant adopted the Clinch River site ground motion hazard curves for the site-specific design response spectra, which you saw in the applicant's slides.

Again, this is a site that's 3.5 miles from the Clinch River site. They are both rock sites with similar geology.

This is all based on the NRC's approved seismic source model, which can be found in NUREG-2115, and also the previously NRC-approved Central and Eastern U.S. Ground Motion Model in EPRI 2013.

In this case, the applicant increased the Clinch River hazard curves by scale factors of 1.5 to 1.7. And this is to account for use of the older ground motion model and the potential impact of site-specific site response analysis.

Next slide, please. The staff's evaluation determined that the applicant's use of the Clinch River hazard curves for the site-specific DRS for the Hermes test reactor is appropriate at the construction permit stage.

The staff also notes that for the operating license Kairos will need to update its site-specific DRS to incorporate new NRC-approved Central and Eastern U.S. Ground Motion Model NGA East, and the site amplification factors determined from the result of site response analysis.

And before that, the staff also plans to perform a confirmatory evaluation using the NGA East model and local site amplification.

Next slide, please. And, finally, in 2.5 we also address the geotechnical engineering in PSAR Section 2.5.4. The applicant determined the subsurface stratigraphy and material properties from a series of blinds and trenches at the site.

The applicant determined that liquefaction was not a concern for safety-related structure, and that bearing capacity of the foundation rock is expected to be adequate.

With respect to settlement for safety-related structures, that's expected to be negligible. And that the settlement for non-safety related structures is something that can be controlled through the use of an engineered backfill.

Next slide, please. The staff's conclusions are that liquefaction is not a concern for rock units encountered at the site, and that the liquefaction potential for non-safety related structures, those that are founded on engineered fill, can be deferred to the operating license.

The staff also notes or concludes that the over excavation to a foundation depth where no evidence of karst dissolution is encountered and, again, not something that would be confirmed by the geologic mapping permit condition.

And, finally, that additional characterization of foundation bedrock can be addressed in the operating license application.

Next slide, please. Overall the Section 2.5 regulatory findings are that the NRC staff concludes that based on the information provided, responses, and our own technical evaluation, that the geology, seismology, and geotechnical engineering characteristics provided for the Hermes site in PSAR Section 2.5 is sufficient and meets the applicable guidance and regulatory requirements for the issuance of a construction permit.

And that further information on geology, seismology, and geotechnical engineering can reasonably be left for later consideration at the construction permit application stage -- or, sorry, at the operating license stage because it's not necessary to be provided as part of the construction permit application.

I'll pause for questions and then, Chairman, I'll then pass the presentation over to my colleagues for their discussion of 3.4 and 3.5.

MEMBER KIRCHNER: Jenise, this is Walt Kirchner. I have a couple of questions for you.

MS. THOMPSON: Uh-huh.

MEMBER KIRCHNER: Since there is extensive cross-reference to the Clinch River Early Site Permit activity, just for my own information, did they use the CEUS -- the, I forget the acronym you used, NGA East or whatever, for that Early Site Permitting activity?

MS. THOMPSON: No. The NGA East was not used for the Clinch River site.

MEMBER KIRCHNER: Okay. So, that's more recent.

And does that account, then, for the scaling factor of 1.5 to 1.7?

MS. THOMPSON: I'm actually going to refer that question to Cliff Munson.

MEMBER KIRCHNER: Uh-huh.

MS. THOMPSON: Cliff, do you want to respond to that?

MR. MUNSON: Sure. I'm Cliff Munson, a senior level advisor for siting in NRR DEX.

So, the factors of 1.5 to 1.7 are intended to account for the potential increase in amplitudes, ground motion amplitudes due to NGA East, and potentially a slight increase due to the site response analysis, the site amplification factors.

So, they increased, they used those factors to account for both of those items.

MEMBER KIRCHNER: Okay, thank you.

And then a last question. Now, looking at the information in the PSAR and then contrasting that with the Clinch River site, what I remember of the Clinch River site is pretty much sitting on this dolomitic rock at the old Clinch River breeder reactor foundations, essentially.

Here it seems like it's on soil on rock.

So, I guess that's -- so, you're permitting condition then is to get a better map of what's underneath so that you can, let me see, ensure that the foundation and et cetera is, is -- I'm searching for the right words -- but could you just explain a little bit more about that permitting condition?

What does that entail in terms of activities: is that more borings and such to answer Dennis' question about karst and such?

MS. THOMPSON: So, I'll give, I'll start with the response and then I'll ask our geotechnical engineer Amit to tag in as well.

The purpose of the geologic mapping permit condition is to look at the clean surface that's already been excavated. So, this would be the actual foundation surface as it's exposed during the excavation process.

And we can go back a couple slides if we want. But the, the intent is to see the removal of the soil and weathered rock in its entirety to expose a clean, hard rock surface.

And I'll ask Amit if he has anything to add on that?

MR. GHOSH: Yes. This is Amit Ghosh.

And I reviewed this part. And what we are looking for is was they are supposed to put it on the hard rock, as Dennis said right now. And there will be a concrete pedestal on which the reactor is sitting.

So, the reactor is specifically sitting on hard rock surfaces. But we want to ensure that there is no karst feature inside. And as Dennis said, like the bore holes gives you a very 1-D information on what is below the surface.

I would prefer, I'd like to see, and it's up to the applicant how to come up with that, some kind of a T.V. picture of the ground, like geophysical technique ground penetration radar or something to show that there is no large void. All we need is small void which can affect.

The reactor is sitting on Murfreesboro limestone, which is -- has fractures, rock joints, dipping quite high in the level 60 degree. And those joints, some of them are clefting.

Clearly, some places, you know, if it gets wet it swells. So, that may open up the joints and all, so the stability and the bearing capacity are the concerns here. So, that's why you want to have a very good characterization of what is below the surface so that we have a good confidence that the design will work.

MEMBER KIRCHNER: Okay. So, the expectation, though, is it will basically -- I'll say this because I'm not a geologic engineer -- you're going to sit the foundation mat on hard rock essentially, is what you're looking at?

MR. GHOSH: Right. Right.

MEMBER KIRCHNER: Okay. And --

MR. GHOSH: Hard rock with sufficient strength.

MEMBER KIRCHNER: Bearing capacity. Okay.

MR. GHOSH: Yes.

MEMBER KIRCHNER: Okay. All right. Thank you very much.

MR. GHOSH: Thank you, sir.

MS. THOMPSON: And if there were no other questions for 2.5, I can pass the presentation to Bryce Lehman who will be presenting 3.4 and 3.5.

Thank you.

DR. SCHULTZ: Before we do that, I have a question.

On the design response spectrum where the applicant has done their evaluation and you've asked for an update to that, whether it's the U.S. ground motion model that has now been recently updated by the NRC, that's something that they need to do before they do the design for construction?

And you'll review that evaluation before that happens?

Is that the sequence of events? I'm looking at slide 51.

MS. THOMPSON: So, my understanding is that the expectation is that the construction, or the construction permit application does not need to address NGA East. But the operating license application would need to include an updated DRS incorporating or using NGA East.

But I'll defer to Cliff if he wants to add more to that.

MR. MUNSON: That's my understanding as well, that they will proceed with the design based on their construction permit design response spectrum, and that they will verify that prior to OL -- as part of their OL application we will see that.

MEMBER KIRCHNER: Cliff, this is Walt Kirchner again.

Let me just pull the thread on this a little bit more. Doesn't that put them at risk a little bit?

I mean, we've had problems with other construction projects where you update the seismic curves that, you know, the spectrum curves and such, and then you, you're in a position of backbiting or making major design changes.

DR. SCHULTZ: Thank you, Walt. This is Steve again.

That's what I was looking for in understanding the sequence of events here. I'd hate to see something move forward on the construction design side and wind up at the operating license time that more work needs to be done.

MR. MUNSON: So, from my understanding -- this is Cliff again -- they are proceeding a little bit at risk. That is why they did their scale factors of 1.5 to 1.7 to potentially account for that.

We don't expect the spectra, the design spectra to be that different.

One thing to note, they are siting the Hermes reactor on hard rock, so the amplification is expected to be flat, basically 1. So, we don't expect amplifications due to the upcoming seismic waves since these are very hard rock structures.

So, we believe that we concluded that there is not going to be much difference, if any, between the CT design response spectrum and the OL design response spectrum.

DR. SCHULTZ: So, the factors that they've applied should account for what is anticipated in the reevaluation?

MR. MUNSON: Exactly.

DR. SCHULTZ: Good. Thank you for clarifying that.

MS. THOMPSON: And I guess if there aren't any more questions we'll go to Bryce.

MR. LEHMAN: All right. Thanks, Jenise.

Good afternoon, everybody. My name's Bryce Lehman. I'm a structural reviewer in NRR, Division of Engineering and External Hazards.

I'm going to go through Section 3.4 and 3.5.

So, PSAR Section 3.4 comprises design overview, pretty similar to what the applicant just discussed. Obviously addresses the SSC required to remain functional after an earthquake, and associated with ASCE 43-19, Seismic Design Category 3.

It uses the graded approach. And the DRS is based on Seismic Design Category 3 and site hazard from PSAR Section 3.5.

Seismic response analysis as well as structure interaction will be performed in accordance with AC -- ASCE 4-16.

Next slide, please. The staff evaluation really focused on verifying the appropriate seismic hazard standards to be used, and that there would be -- the reactor building would be designed to the overview in ASCE 43-19 and ASCE 4-16.

Verify that the DRS is properly developed based on site data from Chapter 2.

And then a lot of information on the specific design details, including seismic and other things will be reviewed during the operating license. But they've been properly identified at this stage.

Within a reasonable level of detail and information provided to the issuance of the construction permit with, again, it's been very clearly identified what staff will expect to see in the operating license review.

So, moving on to Section 3.5, this describes the principal structural elements and design of the reactor building. And it summarizes how the reactor building is PDCs 1, 2, 3, 75 and 76. And it really supports or repeats information that's contained in earlier PSAR sections, especially 3.2, 3.3, and 3.4.

Sort of the unique part of this section, I think, is the discussion of the seismic isolation system. So, we put a little bit of focus on that for this section.

Next slide, please. I'll move through these pretty quickly because, one, the applicant has covered them and, two, like I said, they're kind of referring back to other sections in this.

But SDC-1, the SSC importance of safety shall be designed to quality standards. The staff verified the safety-related SSCs will be designed and constructed in accordance with ACI 349 and AISC N690.

For PVC-2, SSCs important to safety shall be designed to withstand natural phenomena. Again, the structure will be designed to withstand the appropriate natural loads. And there's more discussion about it in earlier sections.

And in PVC-1, structure of the design in accordance with the appropriate nuclear codes, ACI 349 and AISC N690.

DR. SCHULTZ: Bryce, am I missing something. I thought we determined this morning that we are using safety-related and non-safety related. And here we have components important to safety.

MR. LEHMAN: Yeah, I apologize. If you guys had that discussion this morning, I imprecisely said it. I would say unless they've got something to add, if you guys agreed on something this morning, assume that advisement.

DR. SCHULTZ: Yes.

MR. LEHMAN: Okay. All right. Okay.

DR. SCHULTZ: Thank you, Bryce.

MR. LEHMAN: Yeah, thank you.

All right, thanks for that comment.

For PVC-3, the SSCs shall be designed and located to minimize the effects of fire. The reactor building uses low combustible materials and physically separates SSCs. And then, also, there's a fire protection program provided.

And that's discussed in more detail in Section 9.4.

PVC-75, the reactor building shall be protected down to the decay heat removal system located inside the safety-related portion of the reactor building, which provides assurance that it will be protected from the natural phenomena.

And then, finally, PVC-76, reactor

building shall be designed to permit periodic inspection. That is the plan of the design, including the seismic isolation system that will be designed to allow access and permit inspection.

So, for the seismic isolation, as the applicant went over the safety-related portion of the reactor building will be seismically isolated, at this point minimal information was provided on the design of that other than kind of noting that it's going to, going to be there.

They did explain that there would be a moat, as they highlighted during their presentation, which separates the seismic isolated portion from the rest of the plant.

And the system will be accessible and inspected throughout the life of the plant.

So, really here the staff I think have verified that the system will be designed in accordance with the appropriate guidance, which is Chapter 9 of ASCE 43-19. And really highlighted that, obviously, they're going to need a lot more details of the system and the analysis, which will be provided for the operating license application.

Next slide. The conclusions. Safety-related portions of the reactor building will be designed to appropriate codes to support those PVCs.

The seismic isolation system will be designed to the appropriate guidance in ASCE 43-19.

And specific design details that will be reviewed during the operating license have been properly identified.

For the regulatory findings for Section 3.4 and 3.5, the staff finds that there is reasonable assurance that the preliminary information is consistent with the applicable criteria in NUREG--1537, and the relevant PVC, and conforms with 10 CFR 50.34(a)(4), thereby providing preliminary analysis and evaluation of the design performance of SSCs.

Staff also concludes that the information in the Hermes PSAR Section 3.4 and 3.5 is sufficient and meets the applicable guidance and regulatory requirements in accordance with 10 CFR 50.35, and that further information can reasonably be left for the operating license application.

And that's my last slide and I think the last slide of this, this section of the presentation.

Obviously, if there's any questions.

MEMBER HALNON: Bryce, this is Greg.

On these ASCIs, and I think that's the right thing, it's these seismic codes, are those similar or the same as we were using for large light-water reactor post-Fukushima? Or are these specific just for reactors, these non-power reactors?

MR. LEHMAN: Yeah, I'll go ahead and try. And then, Cliff, if you want to jump onboard and add to it.

I think it is the same as sort of the new, the new --

MEMBER HALNON: We're not using then --

MR. LEHMAN: -- revolution of it.

MEMBER HALNON: -- all that we learned after post-Fukushima, and we're using the same, the same methodologies I guess?

MR. LEHMAN: Yes.

MEMBER HALNON: So, when we're reviewing the operating license we should expect to see a level of detail similar to what we saw for post-Fukushima? Or is it going to be a little bit less or --

MR. LEHMAN: I think, --

MEMBER HALNON: -- a lot less, or?

MR. LEHMAN: -- obviously, a lot more than what we've seen here today. About the same as what we would expect for a power reactor. Right?

As far as that level of detail, maybe, Ed, you can support a little bit.

I think, obviously, the staff is going to try, so where that's going to be I'm unsure, but --

MEMBER HALNON: But just curiosity.

MR. LEHMAN: Yeah.

MEMBER HALNON: I mean, I'm just trying to prepare myself.

MEMBER KIRCHNER: Greg.

MEMBER HALNON: Yes?

MEMBER KIRCHNER: Greg, this is Walt.

That suite of codes that were just cited on this last few vugraphs are the same codes that NuScale used to design their reactor building systems.

MR. LEHMAN: Oh, I didn't know that.

MEMBER HALNON: Okay.

CHAIR PETTI: And this is new in terms of regulations compared to, say, 20, 30 years ago. So, it's an evolution. It's going to be positive, I think, as you'll see in the design.

MEMBER HALNON: Yeah. The reason I was asking, you know, my depth of knowledge goes just post-Fukushima large light-water reactor. And I was just curious how, how much harder it is to get to this point, or is going to be the same thing that I'm used to seeing relative to codes?

I describe it as a sharper --

MEMBER HALNON: Sharper. From the light-water reactor?

CHAIR PETTI: Yes. Because these response spectra allow you to propagate things differently than in the old days because we just didn't have the knowledge and the capability of the codes to protect that. Then you put the seismic isolator.

MEMBER BALLINGER: These, the ASCE, the concrete code has been around since --

MEMBER HALNON: Before you were born?

MEMBER BALLINGER: No. They just keep evolving. And the same for these, these folks.

And I actually went and got these codes and ran their numbers. They check out. And they're the same way you would do it for any light-water reactor.

MEMBER HALNON: Okay. All right.

MEMBER BALLINGER: They don't make a distinction, at least --

MEMBER HALNON: Well, that was the, that was what I needed to hear, that it's going to be these similar to what I'm used to seeing.

MEMBER BALLINGER: The reg guides that call out those things are the key; right?

MEMBER HALNON: Well, and this was

completely after the reg guide, so just all going to codes. So that's why I was trying to -- I think I got what I was asking.

MR. LEHMAN: Any other questions?

DR. SCHULTZ: This is Steve Schultz. Just one, Bryce.

It's hard not to miss that the Kairos presentation had within it that the control room is non-safety related. And so, the buildings where that is located is non-safety related, as well as these other buildings onsite.

So, those buildings are just built to the international building codes and the Tennessee Building Code?

MR. LEHMAN: That's correct, yes.

DR. SCHULTZ: All right. Thank you.

MEMBER BALLINGER: I mean, they're relying on the fact that safety, it's passively cooled, so there's nothing that can happen. So, the control room could disappear.

DR. SCHULTZ: Yeah. I understand that. It's just different.

MR. LEHMAN: Yeah, yeah. Exactly right.

DR. SCHULTZ: And good to see. Thank you.

MR. LEHMAN: All right, thanks a lot. I

guess, I don't know if you have any other comments. Other than that, I think we're done now.

CHAIR PETTI: Okay. I think then we can turn to the memos.

Oh, there's a hand up? Yes, Kairos, please.

MR. PEEBLES: We had a couple of corrections to make, so I'm going to hand it over to Brian Song again.

MR. SONG: Yes. It's Brian Song.

So, I think the first item is about the karst. In our slides we indicated no karst information. However, we did an investigation throughout the whole site, and we did discover that the karst information is north of where we located the Hermes site.

So, that is kind of what we tried to indicate in the slides. So, hopefully that clarifies that.

The second item is the question about the tools being the same of the safety-related and non-safety related structure that where it's located. Based on PSAR, we are, for the non-safety related items we are using ASCE 7 and building codes to, to analyze those items.

So, the tools may or may not be the same. So, I just wanted to clarify that as well.

DR. SCHULTZ: Thank you.

MEMBER BROWN: Dave, this is Charlie. One question not explicitly related, but sort of.

Main control building is separate, it's non-safety related.

There are some things that I'll bring up in Chapter 6 as well as 7, in that there's some manual controls associated with actuating or ensuring things have actuated in the DHRS and the reactor trip type functions. So, if they're separated and it's non-safety, it would be interesting to at least address how we maintain those remote on backup shutdown capabilities for those two safety-related issues.

That's all. Just put that in the hopper to talk about whenever we get to them.

CHAIR PETTI: Yeah. And I had the same thought with DHRS actively engaging.

MEMBER BROWN: Yeah, there are a couple of things that are -- we'll talk about some of those tomorrow when we talk about Chapter 6, so.

CHAIR PETTI: Okay.

MEMBER BROWN: DHRS is obviously the one. But there's also a water source for the water tanks and things like that that are -- feed the tanks, even though it's in a safety-related area it's got sources. And how do you control that source?

Just another point that we didn't talk about.

CHAIR PETTI: Yeah. I think that's best talked about tomorrow.

MEMBER BROWN: Pardon?

CHAIR PETTI: We'll talk about that tomorrow.

MEMBER BROWN: Yeah, yeah. I just wanted t make sure we had, because it was remote and it's non-safety, therefore, and there are controls, and it's not like everything else, so I just wanted to get it on the table so people have it in their brain, in case I forget.

That's it. Thank you.

CHAIR PETTI: Kairos again? No? Or do we take a break?

MR. PEEBLES: Yeah. I was just going to preview. So, we will discuss that more in Chapter 6 and Chapter 7. But the backup shutdown functions are not safety-related.

So, the primary shutdown functions in the reactor protection system are what's safety-related.

But we can talk more about that in the Chapter 6 and Chapter 7 discussion.

CHAIR PETTI: Okay. So, let's, we have Chapter 3. Should we do Chapter 2 first? Either way?

(Whereupon, the above-entitled matter went off the record at 1:39 p.m. and resumed at 2:01 p.m.)

CHAIR PETTI: Kairos, you're up.

MR. LATTA: Hello, my name is Ryan Latta. I'm a principal engineer here at Kairos Power, in the fuels and materials area.

I've been at Kairos for five years now. I previously presented the fuel qualification methodology topical report, that was reviewed and approved just this month.

And now today I'll be speaking about the Hermes PSAR, Section 4.2.1, on reactor fuel.

Next slide, please. So this figure here just talks to the fuel form. I believe it's made it's rounds and been seen at this point.

We use the TRISO fuel particle that's within a pebble fuel form. The TRISO fuel particle is based on the AGR, DOE AGR program.

The TRISO particle has the kernel and multiple layers, that are part of our functional containment.

The particles lie within the fuel form, which is a pebble. The pebble has fuel, three regions. An inner density, or inner, inner core that's of a low density to low buoyancy of the pebble.

The fuel region that contains the TRISO particles, and the outer fuel free shell that separates the TRISO particles from mechanical, or chemical interaction with materials outside of the pebble.

The carbon matrix is the same material throughout the fuel pebble.

In the core, besides the fuel pebble there are also moderator pebbles. These are homogenous pebbles made up of the same carbon matrix material, with the same diameter. They do not contain fuel.

And the purpose of the moderator pebbles, is to get the right carbon to heavy metal ratio, so that you have optimum moderation within the Hermes reactor.

Okay, next slide, please. Okay, this just further talks to a description of the fuel. The main kind of parameters of the fuel particle are shown on the left, with nominal values. These values are consistent with AGR-2 and AGR 5/6/7 fuel specifications.

The description of the pebble on the right, shows the dimensions and overall densities, which are the same for either the fuel pebble, or the moderator pebble. And then of course for the fuel pebble, the loading information.

Next slide. Okay, this slide talks to our fuel qualification program. And so as I said previously, we had reviewed, had reviewed the fuel qualification methodology topical report, and I'll give a kind of quick outline of what that entails. It's described in the fuel summary.

So, the foundation of our fuel qualification is the DOE AGR program. We used an equivalent fuel specification for our fuel particle, that we use in Hermes.

And we rely on the EPRI topical report, that demonstrated AGR-2 irradiation safety performance that showed good performance of the fuel to low failure fractions for that fuel.

And, we're leveraging that irradiation safety testing experience, along with that design in Hermes.

And in addition to that, we performed a PERT exercise looking at fuel particle and pebble phenomenon, in our application with a figure of merit.

And through that, we identified high priority phenomenon that we would investigate further, as part of the fuel qualification program.

This ended up centering on the development of a pebble laboratory test program, where the purpose was to demonstrate that the fuel pebble would meet the functional requirements in the reactor.

Part of that includes all the non-irradiated testing in a laboratory, such as mechanical testing to demonstrate pebble structural integrity, geology to look at where within different environments of its, in its Service life.

Molten salt filtration to investigate buoyancy of the pebble, and then material compatibility in the environment that the pebble would see in it's Service life.

And looking at interaction between the pebble and the environment where there's salt, or air environment.

Okay, next slide here is the fuel qualification envelope. So we base our, I said before we base the qualification of the TRISO particle on the EPRI topical, and that looks at AGR to irradiation and safety performance.

So, we used that information to create our fuel qualification envelope. Here we identify four specific parameters of temperature, burn up, power, and fluence.

And then defined values for those for normal operations, or accident conditions. And then for the design of Hermes, we operate within that envelope as part of our qualification.

Okay, next slide. So with the operation of Hermes, there is a fuel surveillance program, and this includes a couple parts.

The first part is monitoring the coolant and pepper gas, for radioactivity. In this instance, we would be looking at trends and increases in activity that could be associated with fuel particle failures, to indicate that the, the operations and kind of health of the fuel.

Furthermore, then there's the second part, which is inspection of fuel pebbles, and the pebble handling system after pebbles exit the core.

So the pebbles go through the core multiple times, as you know, and once they exit the core each time, they would be examined for gross damage, and, and burn up, to demonstrate that the pebbles are still within the operating envelope for burn up.

If pebbles exceeded the operating envelope, or exceeded the burn up limit, sorry, or exhibited damage, then they would be removed from Service and placed in storage versus being recirculated through the core.

This speaks to the design bases. There are two PDCs addressed in the PSAR. PDC 10, related to direct core design, and PDC 16, related to the functional containment for PDC 10 as a sub-core.

We're relying on the AGR particle design, and the every topical design or qualification envelope for our fuel, that then the Hermes reactor must operate within.

Speaking to functional containment or PDC 16, the TRISO particle makes up a large portion of the functional containment.

There are four barriers within the TRISO particle. The kernel, and brief coating layers that prevent the release of radio nuclides.

Further protections are provided by the pebble inspection system, that examines the burn up and the physical condition of the pebble, and the effort to remove pebbles that are damaged, or exceed burn up limits.

And then for the pebble form itself, we

have a pebble laboratory qualification program, where we through this series of mechanical and other chemical tests, to demonstrate that the pebble meets it's functional requirements, and protects the TRISO particles from damage.

CHAIR PETTI: Ryan, I had a question on a number. You assume, I guess it's going to be a limiting condition for operation, on failure of 2.310 to minus 3.

The table says it's a SiC failure, but I'm, want to understand from the model from the source term.

Do you assume the OPIC has also failed when you look at that number?

MR. LATTA: Yes. So those are a combination of manufacturing defects, and in-Service failure fractions.

Those SiC failures are most related IPIC failure that's occurring at a higher fraction.

CHAIR PETTI: Right.

MR. LATTA: Yes.

CHAIR PETTI: Yes, because of the lower temperature.

MR. LATTA: Yes, primarily, yes. So there's a high failure fraction of IPIC, due to the lower temperatures that we see in the models.

And, that SiC failure fraction is driven by those IPIC failures.

CHAIR PETTI: I just want to make sure as I capture it properly, when I talk about function containment.

MR. LATTA: That's correct.

CHAIR PETTI: Great, okay. Thank you.

Other question. Have you looked at all about systems to inspect the pebble, as it's rolling in the handling system?

Have you looked at cameras and the like, and the doses? I just say talk to the guys in Idaho because they've done a lot in the, in-cell.

Pictures are amazing, but they burned out a lot of cameras because of the dose.

MR. LATTA: Yes.

CHAIR PETTI: So there's a trade off there. So if you want you know, really good resolution, you've got to get closer and there's that trade off.

There's a lot out there compared to say 20 years ago.

MR. LATTA: Yes, I appreciate the comment and we're definitely prototyping systems to, to you know, develop a full system for Hermes.

MEMBER BALLINGER: Yes I mean the inspection techniques should be, this is consistent with the expected fraction of failure.

If you got an inspection system that can see 10 to the minus eighth, that cost, cost $100 million, that's not going to do you any good if all you need is 10 to the minus four.

CHAIR PETTI: No, I'm more worried about physically, I mean they're going to look for damage. So you know, what's the best, how much can you see? What's the resolution?

MEMBER BALLINGER: Yes, but there's no reason to be able to see such detail when it has nothing to do with the site.

CHAIR PETTI: If it, yes.

Go ahead, Ryan, keep going.

MR. LATTA: That concluded the presentation.

CHAIR PETTI: Oh, that's right, 4.2 is it's.

MR. LATTA: 4.2.2 is a separate section.

CHAIR PETTI: Oh. No, it's discussed now, just a different person.

DR. DORON: Are we ready? Yes, okay.

CHAIR PETTI: Yes.

DR. DORON: Hi, this is Oded Doron, the senior director of reactor system design, and I'll be talking today about the 4.2.2, the reactivity control, and the shutdown system. Or as we call it, the RCSS.

Next slide. RCSS is composed of two systems. The reactivity shutdown system, the RSS, and the reactivity control system, the RCS.

The RSS is credited for reactor trip and shutdown. There's three safety related shutdown elements, that insert directly into the pebble bed.

The reactivity control system or RCS, is inserted on reactor trip, but not credited. There's four non-safety related control elements that are inserted into the reflector.

The release mechanism is a safety related electromagnetic clutch. The drive mechanism is a non-safety related motor driven sheave, to position elements. This provides for position indication.

Testing and inspection. The RCSS is periodically inspected for wear. The reactor coolant is periodically sampled for an increase in boron concentration, that could indicate shutdown element cladding failure.

The RCSS elements can be replaced, if necessary.

On the right there, a simple diagram showing the four RCS elements inserting, or where their locations would be in the graphite structure, the reflector structure. And, the three cruciform RSS elements in the bed.

Next slide. So the shutdown elements are cruciform design. The inner cladding contains the absorber, B4C. There's an argon fill. The cladding is stainless steel 316H.

And on the right there, you can see a simple diagram again of the design. And unless there's comments, I'll just go on.

The control elements are segmented annular design. You can see the diagram on the right there.

There is individual capsules. Again argon filled, the absorbers B4C, and the cladding is stainless steel 316H.

CHAIR PETTI: So just a question. Are there any concerns on temperature limits? You know, they're going into some pretty hot locations, and usually metals in the high regions, you might get close to limits.

DR. DORON: I missed the beginning, the start of your question.

CHAIR PETTI: Just the, no, the

temperatures being experienced by the control elements, and even the shutdown elements --

(Simultaneous speaking.)

DR. DORON: Yes.

CHAIR PETTI: -- relative to the ASME allowables on the --

DR. DORON: Yes, they, well yes, that's being considered. And, they are going to stay within the allowable.

CHAIR PETTI: Have you done the calculations to know that? But you just?

I've just, I've been involved in a number of designs. This is an area that we just have to look at carefully.

DR. DORON: Yes. The details --

(Simultaneous speaking.)

CHAIR PETTI: I mean with Flibe, you may get a very big different temperature distribution, than in gasses.

DR. DORON: Yes, uh huh. Yes, I mean the detailed analysis is going to come in OLA, but that is a requirement.

MEMBER BALLINGER: This, I didn't read about this, but this is Ron Ballinger. Do these, does this plant operate with all rods out?

DR. DORON: Yes.

MEMBER BALLINGER: Okay. So it's not in, the rods are not in the hot --

(Simultaneous speaking.)

DR. DORON: No.

MEMBER BALLINGER: -- section unless it's really --

CHAIR PETTI: The shutdown rods are out obviously, but the shutdown --

(Simultaneous speaking.)

DR. DORON: The controller -- I'm sorry.

CHAIR PETTI: Go ahead, go ahead, please.

DR. DORON: The shutdown elements are fully withdrawn during operation.

MEMBER BALLINGER: But the control elements?

DR. DORON: They may be inserted depending, but during steady state, they'll be almost completely withdrawn.

CHAIR PETTI: But this reactor will never really get to steady state. It won't get to equilibrium in four years, most likely, at the capacity factors.

Just something we're going to have to watch.

MEMBER BALLINGER: Anyway, the thermal load is not likely to be as if everything was in. You know, if they were in the hot section all the time.

CHAIR PETTI: All the time, but they'll be in for some fraction of time.

DR. DORON: I'm sorry, it's difficult for me to hear the second gentleman that was talking. I couldn't hear if there was a question there.

MEMBER REMPE: Ron, you need to turn on your mic.

CHAIR PETTI: Yes, Ron, turn on your mic.

MEMBER BALLINGER: No, no, just me rambling.

(Laughter.)

DR. DORON: That works, too.

Okay, all right, there's a lot of words here, I'm just going to read through them. Stop me if I bore you.

The safety related RSS is capable of operating during an earthquake. The insertion capability is confirmed via testing and maximum deflection of insertion path, due to an earthquake. This satisfies PDC 2.

The RSS is compatible with environmental conditions, and confirmed by qualification testing.

Analysis demonstrates internal gas pressure due to irradiation, does not exceed safety related RSS element stress limits. This satisfies PDC 4.

RSS is designed to fail in a safe state when the plant trips, or upon loss of normal power. The energy holding relay is closed, to remove power supply holding shutdown elements in place.

And, the loss of power allows shutdown elements to drop via gravity. This satisfies PDC 23.

The RCS, remember this is the RCSS, which is the RCS and the RSS together, meet PDC 26. This is discussed in Section 4.5, the PSAR nuclear design.

The RCSS is designed to limit the amount and rate, of reactivity insertion by controlling the maximum withdrawal speed of control and shutdown elements. This satisfies PDC 28.

The design of the RSS trip function, in conjunction with the reactor protection system, assures an extremely high probability of accomplishing its safety related function.

Both the RSS and the RCS, provide significant negative reactivity insertions of the core via gravity and motor driven means, upon a reactor trip. This satisfies PDC 29.

CHAIR PETTI: I have a question sort of

topicologically. The motors and the clutch, you know, the top of the reactor there's going to be some shine off the top because there's not a, you know, an upper reflector.

And in some designs I'm aware of in the past, the motors, the radiation damage issue and combined with the temperature, means you can't use like a light water reactor design option.

Have you guys looked at that? Is there going to have to be some testing, or are they far enough away that they're protected?

DR. DORON: It's a combination of things. The details will come in OLA, but what I can tell you is both temperature and radiation effects are taken into consideration, indeed.

MEMBER KIRCHNER: Can I follow on, Dave? This is Walt Kirchner.

DR. DORON: I wanted to add one, just a bit of a clarification.

While the thickest part of the reflector is certainly on the side, there is still graphite above the core. There is fuel in the fueling chute, but there is still graphite above the core.

And, so there is some radiation protection that is provided by the graphite, and also our head.

While the dimensions have not been provided, I will say is relatively thick and provides some radiation protection, as well.

All that to say again, thermal and irradiation effects are considered in this design.

MEMBER KIRCHNER: May I ask a question? The shutdown system, that's going to go in by force of gravity, right?

DR. DORON: Correct.

MEMBER KIRCHNER: Yes, so do you have a scale test or something in mind to demonstrate that that cruciform rod will penetrate the pebbles, and is, has a sufficient weight such that will overcome the up flow that exists in the core?

DR. DORON: Yes, so we've done significant testing already, but we have --

(Simultaneous speaking.)

MEMBER KIRCHNER: Okay.

DR. DORON: -- committed to a test for the PSAR specifically, to demonstrate that.

MEMBER KIRCHNER: Excellent, okay.

And, then it wasn't clear from the design details that were available in Chapter 4. Is there a positive way to drive those rods in?

DR. DORON: Yes.

MEMBER KIRCHNER: It seems like it's a cable clutch arrangement, but I didn't understand or maybe I just didn't, I missed the detail.

DR. DORON: There is not --

(Simultaneous speaking.)

MEMBER KIRCHNER: Is there a way to do a positive insertion?

DR. DORON: There is not a safety related design to do forced insertion besides gravity, for the shutdown elements.

MEMBER KIRCHNER: Okay.

MEMBER KIRCHNER: But are they capable of running up?

DR. DORON: Yes, it's capable. It's capable, yes. But we're not committing to it.

MEMBER KIRCHNER: And, could you just briefly describe how that's accomplished? Because it wasn't clear from PSAR.

I mean you've got a hoist-like system.

DR. DORON: Yes.

MEMBER KIRCHNER: It wasn't clear to me how you got positive insertion.

DR. DORON: Yes, but through the hoist system, I mean that's why I'm saying it. I would not commit to saying that we would shove in a sense, the elements in with it.

But we have a means of moving the element up and down, if absolutely necessary. But there's no commitment to it.

MEMBER KIRCHNER: Okay and again, but that would just be gravity as the, as the element --

(Simultaneous speaking.)

DR. DORON: Correct.

MEMBER KIRCHNER: -- was, the winch went in the opposite direction?

DR. DORON: Correct.

The idea here with the shutdown elements especially, is extreme simplicity. What we want is gravity insertion.

MEMBER KIRCHNER: Yes.

DR. DORON: And we want almost nothing else to be able to stand in the way of that gravity insertion.

MEMBER BALLINGER: Yes, this is Ron Ballinger. There's quite a difference between a dry bed, which is what the German pebble bed was, and a wet bed, which is what this is.

In other words, there's molten salt in there. So, it's not a dry bed where the rods have to jam down through there.

It's kind of a well, let me get my way through this kind of thing with a lubricated bed.

DR. DORON: Yes.

MEMBER KIRCHNER: Yes, so my, one of the concerns I would have is just I'm sure you've looked at this.

As you described, there is an upper reflector. And as I understand it, that whole space is filled with Flibe.

During normal operations, the shutdown rods would be retracted into that space and up into the enclosure attached above the upper vessel head.

At some point, where does the Flibe stop, and do you have to worry about freezing of the Flibe preventing a gravity drop of the cruciform rod?

DR. DORON: That's a very good observation. The elements when I say quote, fully withdrawn, they're sitting in the reflector region just above the core, not above the hot well.

The tips are sitting below the hot well. So they're sitting right above the core region. Therefore, Flibe freezing wouldn't affect our insertion.

Does that make sense what I just said?

MEMBER KIRCHNER: Yes, as long as you

didn't get cold spots or something, you know, as a result of tight clearance and tolerances.

DR. DORON: I mean, yes, but I guess that would need to be considered, but I don't see a physical way for that to happen in the transient time that we have.

MEMBER KIRCHNER: Thank you.

DR. DORON: Yes.

CHAIR PETTI: And just to confirm, as I recall reading the piece, there was no need for really rapid shutdown.

DR. DORON: That's correct.

CHAIR PETTI: These are like all gas reactors. They can take, it would be slower as long as they take their own time.

DR. DORON: Yes, you're correct.

CHAIR PETTI: Yes.

DR. DORON: This was my last slide.

CHAIR PETTI: Okay.

DR. DORON: I appreciate y'alls time, thank you.

MEMBER KIRCHNER: Dave, this is Walt.

CHAIR PETTI: Yes.

MEMBER KIRCHNER: Sorry to be so pesky today. I have another question.

It's a simple one, but I'm curious how competent you will be about your fueling system, and when you put the moderator pebbles into the reactor, how do you know where they are distributed vis-a-vis, the fueled pellets?

I assume that they'll, each type pellet or pebble, sorry, will have the same mass. But what's the, do you have any uncertainties as to where those moderator pellets are, pebbles are, such that you might get hot spots?

Or is it just statistically --

(Simultaneous speaking.)

DR. DORON: Yes.

MEMBER KIRCHNER: -- pretty random?

DR. DORON: I don't want to punt the question per se, but what I'd ask is coming up next is some nuclear design.

Our nuclear design manager is going to be talking. He'll be touching a little bit on the start up sequence.

MEMBER KIRCHNER: Yes.

DR. DORON: Then there will be a presentation I believe on the force on our pebble handling system, as well. Is that correct? Tomorrow, okay.

I would ask maybe let's go through those presentations.

MEMBER KIRCHNER: Okay.

DR. DORON: And then let's revisit --

(Simultaneous speaking.)

MEMBER KIRCHNER: Yes, that's fair enough.

DR. DORON: Yes, and let's revisit that question after that. I think it will be better than me trying to talk at a blank screen right now.

MEMBER KIRCHNER: Okay, thank you.

DR. DORON: Okay, thank you.

CHAIR PETTI: So, let's then try to get the staff in before the break.

(Pause.)

MR. SCHMIDT: Hi, I'm Jeff Schmidt, Reactor Systems and Advance Reactor Division. I did the 4.2.1 fuels review. And with me is Ben Adams. He did the 4.2.2 and 4.2.3 reviews.

Go ahead, next slide. So this just described what I just said. So the 4.2.1 is the fuel, 4.2.2 is the reactivity control systems that Kairos just went through, and 4.2.3 is the neutron source.

Go ahead, next slide. So 4.2.1, Kairos went through this as well, I'm going to go through it fast. Obviously, the TRISO particle, use of a larger

AGR-2 fuel kernel as I described.

Nominal AGR coating thicknesses, arranged in an annulus as they described. Smaller size than typical HTGR pebble size.

And then this also, this section also covers the moderator pebbles, the non-fueled pebbles as I described, and covers the fuel performance monitoring they also described.

Next slide. The regulatory basis preliminary safety analysis report, issuance of a construction permit, and common standards. We've seen that in multiple slides here.

The guidance we used was NUREG-1537 again. Principal design criteria. Sure, the principal design criteria for 1016 34 and 35 considered.

Next slide. And this just lets out that topical report, some of which we described already. Principal design criteria, we've described.

They also described the EPRI topical report, which is now the foundation for the TRISO particle that they're using.

Fuel qualification, we also discussed this morning. This deals primarily with the pebble protecting the TRISO particle, and the various tests they're going to do.

And then one we didn't mention but we talked about previously, was their fuel performance methodology, which is going to use KP-BISON as their fuel performance code.

Next slide. Particles will be within the UCO particle parameters that they have in their topical report, table 5-5, EGR irradiated conditions found the Hermes expected normal operating conditions.

Expected possibly at an event the temperatures are below the 1600 C; the lowest AGR program safety testing temperature.

An AGR test program did not include overpower transient test and we'll describe that, or discuss that in the following slides.

Pebbles, the fuel pebbles. The actual pebble material, the outer matrix is not credited as a fission production retention.

The topical report, the fuel qualification topical report, addressed pebble testing. Specific correlations if necessary, will be derived from the test data and applied in the Hermes FSAR.

So things like pebble oxidation rates, their testing. They may use different ones in the final design, based on their testing relative to what they used in the PSAR.

Moderator pebbles contained no fuel, and is a non-safety related component. Same material as the fuel pebble matrix.

Testing for buoyancy, wear, strength, salt, ingress will be the same as the fuel pebble described in the fuel qualification topical report. The PHS will inspect the fuel pebbles, and also the moderator pebbles.

The staff, the staff's review for the moderator pebbles focused at least from a fuel standpoint, focused mostly on the impact of any potential debris generated that could impact safety functions, like the fluidic device. And, the natural circulation flow paths.

So it was almost focused on, at least from this aspect, debris generation and any impact because it's a non-safety related component.

MEMBER KIRCHNER: Jeff, this is Walt.

MR. SCHMIDT: Yes.

MEMBER KIRCHNER: Sorry to interrupt again.

MR. SCHMIDT: Sure.

MEMBER KIRCHNER: Could you elaborate on that last sub-bullet, because why wouldn't that also be an issue for the fuel topical?

MR. SCHMIDT: It is for the fuel but you

know, they have, as outlined in the fuel qualification topical report, they have extensive testing to make sure that that does not happen, right?

MEMBER KIRCHNER: Okay.

MR. SCHMIDT: What wasn't described in the fuel qualification topical report is, what are you doing for the moderator pebbles.

So, the staff wanted to make sure that the moderator pebbles and its effect on the safety functions, were also addressed.

And that's why the PSAR brings that specific issue to light.

MEMBER KIRCHNER: Okay, thank you.

MR. SCHMIDT: So the AGR test program performed relatively slow heat ups, like those experienced for a loss of flow type events.

It does not really cover the Hermes overpower transient heat up rates, which are significantly faster than the AGR tests.

Overpower events can potentially lead to fuel melt, or non-melt mechanical failures. Limiting overpower transients are generally rod injection, and rapid element withdrawals.

Rod injection is precluded by design, and this event is a low differential pressure that was discussed earlier today.

Staff evaluated the maximum control element withdrawal, presented in Figure A1-2 for the postulated event analysis methodology, the technical report KP-TR-018.

So that was, we're using that information to inform our review of whether overpower transient would be necessary, overpower transient testing would be necessary for the Hermes testing.

It gives us an idea of what type of rapid positive reactivity we could be having.

Next slide, please. So here are some specific numbers. The maximum control element withdrawal temperature is 1080 C.

This is well below the unirradiated UC temperatures 1.96 and 1.86 of 1800 C, and 2350 C. And I just point out that the AGR program, they really only identified UO2 UC and the UC 18.

So the 2350 is the, probably the most appropriate temperature out of those. I just listed the other ones as the lower bound.

When I did literature searches, the 1.96, UC 1.96, had something down to 1820, but it was not identified in the AGR program.

The energy deposition for this transient, for their transient, is a complete control element withdrawal, was significantly below the 1400 Jules per gram for UO2, for the failures identified, and for fresh UO2 testing that has, was done in Japan.

Overpower event non-melt failures are also expected to be negligible. The time to reactor trip is approximately 10 seconds, so the event from, from withdrawal to trip is 10 seconds.

The fuel thermal time constant is between 30 and 300 milliseconds. And over that time period, basically the energy has time to escape the particle and get into the environment.

NC wouldn't expect melt or even non-melt failure modes to occur.

Let's go to the next slide, please. And this is the graph of that. So basically the delta t across the particle versus the energy deposition duration, for a range of Jules per gram.

And you can see, you know, we're talking like a 10 second evolution. You can see there's almost no delta t across the particle that would be driving potential failure modes.

And that's really an important aspect of how the staff got to the conclusion that overpower transient testing is not necessary for, for, excuse me, for the Hermes design.

I'm dry today. Any questions on this slide before I go on?

No? The vertical line for the dash is (audio interference). Negligible incremental failure is predicted by the KP-BISON.

So, in addition to that graph I just showed you, you know, we also brought in other aspects to determine you know, if overpower transient testing was necessary for the Hermes design.

And these are some of the other considerations that the staff used, in reaching its determination.

You know, we just looked at the KP-BISON results. The KP-BISON is not validated but it is informative. It's not necessarily wrong at this point, so we used that as kind of a means of supporting our argument.

And, that predicted Service incremental failures. There was no difference between the in-Service normal, and the transient induced failures.

If you look at those failure fractions, they're identical. So while we'll look at that again at the OL after they've done the VNP, but right now, you know, we're not seeing any significant incremental failures.

And, that's not surprising based on the testing that has been done in the AGR programs is you know, I know there's slower transients, but you didn't really see any type of significant jump in incremental fails till you got pretty high temperatures from 1800 C, so.

Based on the low predicted maximum TRISO temperatures of this design, the corresponding margin to melt, the low stress caused by the small deferential coating temperatures, and the review of the preliminary BISON results, the staff finds that incremental failures are not expected during the worst case overpower transient events.

And you know, transient, overpower transient testing is not necessary to support the Hermes application.

Next slide. So this is on fuel performance monitoring. That was also discussed by Ryan. Three non-destructive means are available to monitor fuel performance.

Cover for gas, reactor coolant, technical specification reactive coolant as given in PSAR Chapter 14, but specific values will be provided as a part of the OL.

Fuel pellets will be examined for damage and burn up, by the pebble handling system. Pebbles which show abnormal wear, cracking, or surfaces will be removed from Service.

Pellets will be discharged to prevent exceeding the burn up. And that was all discussed prior during our fuel qual topical report review.

Next slide. Destructive testing of the Hermes fuel will be formally stated in the section 3.9.3, of the fuel qualification topical report, KP-TR-011.

Destructive testing will determine failed fuel fractions, pebble wear, and in the extent if any, of Flibe ingress.

Destructive testing can provide additional fuel performance code validation, and provide input to, to revise any future KP-FHR or fuel FIRT, if necessary.

Next slide. Technical evaluation conclusions. Pressure particles are expected to operate within the bounds defined by AGR-2 test data.

Incremental failures are expected to be negligible based on all the events, including the worst overpower transient.

Fuel moderator pebble testing programs are sufficient to develop relevant acceptance criteria correlations, to ensure pebble safety.

Safety functions are satisfied, and the fuel monitoring is adequate to determine unexpected fuel failures to determine those.

I guess I'm just going to pause there before we go on to the 4.2.2 section, to see if there's any questions from the 4.2.1.

That covers a lot of information here.

MEMBER KIRCHNER: Jeff, this is Walt again.

Just a rhetorical question. What if in your last bullet on the preceding slide, the activity detected in the cover gas system turns out to be higher than expected?

Would you then look for destructive testing of pebbles, to make sure there wasn't manufacturing induced defects?

I mean one of the not open questions, but certainly one of the changes in this design from the German pebble, is to go to an annular fuel region with a higher packing fraction.

So substrat you know, the overcoat when you compress and fire the final fuel form, isn't as how should I say forgiving, as in, as in the German pebble bed design.

MR. SCHMIDT: Yes --

(Simultaneous speaking.)

MEMBER KIRCHNER: So given that this is a test reactor, would you then look at the performance, and then look for the applicant to, or well, it would be in the operating mode by that point, do destructive testing?

MR. SCHMIDT: Yes, I think initially, you know, it's going to be governed by some type of tech specs, right.

You start to see activity levels above what you anticipate, you're going to have to take some type of action. And, I guess that action will be defined at the OL as part of the tech spec actions.

You know, what you do from there on to identify what was causing that, I'm not so sure. You know, I'm not sure I know the answer to that.

CHAIR PETTI: That's a needle in a haystack problem.

MR. SCHMIDT: Yes.

MEMBER BALLINGER: But would you --

(Simultaneous speaking.)

CHAIR PETTI: It's a simple game of spectrocity. You won't be able to tell very easily.

MR. SCHMIDT: Right.

CHAIR PETTI: And you're not going to destruct every pebble, you know.

MR. SCHMIDT: Yes.

MEMBER BALLINGER: Would you be able to find out in the pebble handling system if you waited long enough?

In other words, you detect that activity in the bed and it's not, it's not critical in any way, so you just hang on and wait until you see it.

You know that a pebble's got to be failed.

CHAIR PETTI: The handling system with spectrocity is all about burn up.

MEMBER BALLINGER: Yes.

CHAIR PETTI: So they look at cesium. Okay, if you fail the particle, cesium's going to get out. It's going to get into the matrix.

But you're not going to see that most likely, with the spec system, the gamma spec system. It's just going to look like the cesium, yes, it's moved a little bit.

MEMBER BALLINGER: Could something to be done to the handling system to, to allow for additional resolution, if you will?

CHAIR PETTI: The only way that I know that would be a program of SAR failures physically, is in destructive examination.

You knew you had them in radiation from gas activity, but then you couldn't say which one it was until you did something more sophisticated.

MR. SCHMIDT: You would have to do something different. They have not committed to this, so don't get wrong. This is just my idea of like you know, they do sipping for a light water fuel.

You could arrange some type of off gas type sipping arrangement, that may identify what pebbles have significant failures that could go to destructive testing.

But I mean that's just speculation on my part, nothing more.

MEMBER KIRCHNER: Yes, that's what I was thinking, Jeff, that you would have, you said sipping. I would say sniffing in this case.

MR. SCHMIDT: Okay, okay.

MEMBER KIRCHNER: But, yes. Okay, well, this is the advantage of having a prototype test reactor. Okay, thank you.

MR. SCHMIDT: Yes. Any other questions before I move on?

MR. ADAMS: Good afternoon, I'm Ben Adams. I will be going through the 4.2.2 section on the

Hermes reactivity control and shutdown system.

CHAIR PETTI: Either speak louder, or pull it towards you more.

MR. ADAMS: Is this good? Does this sound better?

Okay, I'll start over.

Hi, I'm Ben Adams, and I'll be going through the 4.2.2 section for the Hermes reactivity control and shutdown system.

The regulatory basis is the same as it's been in the previous presentations. It is 10 CFR 50.34(a), 10 CFR 50.35, and 10 CFR 50.40.

The relevant guidance that we used to conduct the review of the section was also NUREG-1537, and the findings in this section are related to same.

The Hermes design is consistent with the principal design criteria, which are listed on the slide here.

That's going to be PDC 2 design basis for protection against natural phenomena.

PDC 4 environmental and dynamic effects design basis.

PDC 23, protection system failure modes.

PDC 26, reactivity control system.

PDC 28, reactivity limits, and PDC 29,

protection against anticipated operational occurrences. All but one of the findings in the section are linked to the PDC.

We can go to the next slide. So this is a bit of an overview slide and Kairos discussed this just right before we did, but I'll go through some of it again.

The reactivity control and shutdown system is credited shutdown, the RCSS and sort of safe shutdown by inserting electronic elements to control reactivity during normal operation and response to normal conditions for postulated events.

There are four control elements, and three shutdown elements, and they all use the B4C absorbent material.

In the PSAR, Kairos committed to periodically inspecting the shutdown limits for wear, and damage to the cladding that encapsulates the B4C.

And they've also committed to monitoring the Flibe coolant for boron content, which would provide the location of the B4C cladding failure, and the control limits and shutdown limits can also be removed and inspected, or replaced.

We are comfortable at the construction permit stage with this level of information that the specific details of the monitoring and performance monitoring requirements will be handled later at the OL stage.

Dan, the other slide, go back one slide. Okay.

At the control elements, they insert into the graphite reflector on the periphery of the core, and they have a range of insertion positions.

The shutdown elements insert directly into the bed, and they should either be fully withdrawn, or fully inserted.

And they are credited with shutting down the reactor, unlike the control elements.

We don't have a picture of them, but Kairos had one in their presentation, of they had different designs and different geometries.

We won't be looking at the differences and the exact design of the geometries at the OL stage, and we'll talk a little why about it in the 4.5 presentation in a little more detail.

Next slide. We kind of talked about the counter-weighted winch system. I won't read through the bullets to describe exactly how it works.

But the PSAR does give a preliminary design description of the counter-weighted winch system.

The release of the clutch is a safety related mechanism and again, we discussed this a little more in our PDC 26 writing, and Section 4.5.

We don't have the exact specific details on what this counter-weighted winch system looks like, like exactly how the rope is supposed to stay in the sheave, or exactly how the clutch releases the elements. We will be reviewing those specific details later at the OL stage.

Go to the next slide. So talk about the findings for the PDC. PDC 22, the design basis and protection against natural phenomena.

Kairos will perform a one-time test before operation that deflects the shutdown on the chi structures by a maximum misalignment that would be caused by a design basis earthquake, in order to confirm that the element insertion time is counted by the insertion time that's assumed in the Chapter 13 analysis.

Staff finds that this is consistent with PDC 2, and we will discuss that analysis, I believe with Chapter 13 on April 18.

Okay, for PDC 4, environmental and dynamic effects design basis. Kairos will perform testing for a shutdown element wear during movement, and will also be expecting analyses for the shutdown element internal gas release, and B4C swelling and other stress limits.

B4C will be monitored like I discussed earlier.

And the, Kairos will also perform analyses for the shutdown elements and control elements, to show that they meet the AMC standards that are discussed in the PSAR.

Staff finds this is consistent with PC 4 and the NUREG-1530 acceptance criteria, that says the RCSS must be designed to withstand anticipated stresses, and the chemical and radiation environment.

Next slide. For PC 23, protection system failure modes, the shutdown limits accomplish safe shutdown via gravity insertion on a reactor trip signal, or on a loss of normal electrical power.

The removal or loss of power, causes the electromagnetic clutch to open.

Staff finds this is consistent with PDC 23, and the NUREG-1537 acceptance criteria.

PDC 26, reactivity control systems, that will be discussed in Section 4.5 later today.

For PDC 28, reactivity limits, the NRC

staff performed its scoping calculations for the transients based on the preliminary information given in the PSAR.

We will discuss these calculations in detail with the Chapter 13 presentation, on April 18.

But they did show that TRISO fuel remains or maintains integrity during a partialated insertion of excess for activity event, and this was used as a primary basis to say that the design is consistent with PDC 28.

And rod injection, we talked a little bit earlier about it, but I think we'll also be talking about it again in the Chapter 13 presentation.

Staff finds this is consistent with PDC 28.

PDC 29, Kairos has planned testing for the shutdown elements, and they have testing for the shutdown element deflection during an earthquake, and shutdown element wear, again discussed earlier.

Kairos will perform periodic inspections of the shutdown elements and coolant, to look for evidence of shutdown element damage and failure, also like I discussed earlier.

And, staff finds this is also consistent with PDC 29.

Next slide. 4D includance. The NRC staff finds that the preliminary design information provided, is consistent with PDC 2 for 23, 26, 28, 29, and it's consistent with the relevant acceptance criteria of NUREG-1537 that's discussed in the safety evaluation.

And the staff has reasonable assurance that the RCSS will perform its safety functions of reactivity control and shutdown.

I believe that's the end of the presentation for 4.2.2.

MEMBER BROWN: This is Charlie Brown. Can I ask a question on the control element reactivity control system?

The wire rope and winch. So, the wire rope, I understand help the drum rotates and the wire rope pulls them out.

But does that mean they can't be driven in? They have to go in just by gravity when you want to unwind the winch?

So there's no force driving these in if you wanted them to, such that if you had a seismic event that jammed the rods, control rods up at the top, they couldn't be driven in? They would be stuck out?

MR. ADAMS: Are you asking about it?

MEMBER BROWN: The wire rope that's kind of flimsy, that's what it seems like. I mean you pull them out, and then you use their weight to drive them back in, to let them sink back in if you release them. As you're controlling them. Incrementally.

MEMBER HALNON: Is there any chance of them binding up and not, gravity not being enough to put them down in?

MEMBER BROWN: No, and you can't drive these in with, with just the wire rope.

MR. SCHMIDT: This is Jeff Schmidt. Yes, that's right.

Their testing though, I think Ben mentioned that they're testing for the full deflection to ensure that they will go in under a design basis seismic event.

So, that testing is addressed.

MEMBER BROWN: So in other words, the sleeve that the control element is in, is wide enough and side-to-side, whatever the dimensions are, that it would absorb and still maintain, an open window all the way up and down, the top to the bottom?

MR. SCHMIDT: Yes, that's right. I think the better way to say it is like the maximum deflection that may occur as it goes through the upper reflector, will be deflected and then ensure you know, like if you get a rapid trip signal, that it can still successfully insert.

MEMBER BROWN: If all the control elements stick out, can the shutdown elements still override and shutdown?

MR. SCHMIDT: Yes, so the shutdown margin we'll talk about in the next section.

MEMBER BROWN: Okay.

MR. SCHMIDT: Just the in-bed rods are sufficient to meet shutdown margin. The control elements, I'm sorry, I said rods, I should say elements. My old habits.

Yes, these are all elements. The control elements just add a defense in-depth. They're not necessary for shutdown.

MEMBER BROWN: So if they didn't unlatch, okay, the drums did not disconnect, whatever, and all four stuck out, or all three stuck out. I've forgotten which ones are which.

Then the shutdown elements going in would override, and still provide the shutdown margin you need. I think I'm trying to phrase that the way you did, or properly.

Is that correct?

MR. SCHMIDT: Yes, I can actually do a little better, too, because it's there are three shutdown elements that go into the bed.

Only two are necessary for shutdown margin because you need the 10 minus one.

MEMBER BROWN: Yes, those are the inside, inside elements, right?

MR. SCHMIDT: Yes, the inside, in-bed elements.

The control elements are in the reflective.

MEMBER BROWN: Okay.

MR. SCHMIDT: They are not credited for shutdown margin in any way.

MEMBER BROWN: Okay, all right.

Thank you.

MR. ADAMS: Okay, let's go to the 4.2.3, for the neutron start up source. This is just going to be one slide.

This section was reviewed using NUREG-1537 again.

For the overview, PSAR does not specify a source type, but it does say that they will have one and that it performs no safety related functions.

This did meet one of the acceptance criteria in NUREG-1537, that says the neutron start up source should be capable of performing in its environment, and removable and replaceable, which is discussed in the PSAR.

We do not know any of Kairos's plans for inspection or monitoring of the neutron start up source. Those will be evaluated at the OL stage.

We will be looking for specific details at the OL, such as source type, exactly how it interacts with the start up plans, and power monitoring, so both of which will be reviewed at the OL stage.

We can go to the next slide. The NRC staff concludes that the preliminary, or this is 4.2.2 slide. Jeff, did you want me to do this, or?

MR. SCHMIDT: Go ahead.

MR. ADAMS: Okay. The NRC staff concludes that the preliminary design provided at PSAR Section 4.2 is consistent with the applicable PDCs and acceptance criteria in NUREG-1537.

The NRC staff finds that the information in Hermes PSAR Section 4.2, is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40.

And, further information can be reasonably left for the OL application.

CHAIR PETTI: Okay, member questions?

DR. SCHULTZ: Yes, just one question to go back to the shutdown element testing.

The two tests that are performed, one is for demonstration that even in the adverse positioning, which might occur during an earthquake, the drop is going to occur. That's one test.

And then also for testing to demonstrate that with wear, the shutdown elements will also perform their function.

Is there periodic testing expected in the technical specifications, that it will be done as the reactor operates?

MR. ADAMS: We do not have those specific details yet, and we'll be looking at that at the OL stage for how that is handled either in tech specs, or some other commitment.

DR. SCHULTZ: When the more detail is available near the operating license. Good.

MR. ADAMS: Right.

DR. SCHULTZ: Thank you.

MEMBER BROWN: Yes, Charlie Brown again.

With the three shutdown elements, if one of them sticks out do, you still have shutdown margin?

MR. SCHMIDT: Jeff Schmidt, yes.

MEMBER BROWN: Even if all the control elements are stuck out also?

MR. SCHMIDT: Yes.

MEMBER BROWN: So you only need two shutdown elements to go in, regardless of the position of all the other elements and control rods, and control elements?

MR. SCHMIDT: Yes.

MEMBER BROWN: Okay, thank you.

CHAIR PETTI: Any other questions, members?

It is 3:00 o'clock. So we are ahead of schedule. We got one more presentation in before the break. So let's break till 3:20 and we'll wrap up today with the Section 4.5. Okay.

(Whereupon, the above-entitled matter went off the record at 3:00 p.m. and resumed at 3:20 p.m.)

CHAIR PETTI: Okay, sorry, Kairos, let's go.

MR. SATVAT: Good afternoon, everyone. My name is Nader Satvat, Senior Manager of Nuclear Design. I will be presenting Section 4.5 of PSAR for nuclear design.

As presented on that image on the right, which is the neutronic model of the core, the reactor core is a packed bed with spherical pebbles. The fuel pebbles contain six grams of uranium. They are enriched up to 20 percent, 20 weight percent uranium 235. Moderator pebbles are used to improve neutron moderation in the core.

The core contains approximately 60 percent pebbles and 40 percent reactor coolant by volume, which is basically the packing fraction of the packed bed. The core is slightly under-moderated, which means negative temperature of coolant and negative void feedback of the reactor.

The general features of the core is that it's continuously refueled. The pebbles are inserting from the bottom of the core. They stay for a period of time within the core region. They're extracted from -- with pebble extraction machine. Go through the inspection and then if decision is made for them to recirculate, they'll go back to the core, up to their design burnup.

The residence time of the pebbles in the core is, at -- for each pass, is about 30-50 days. They're slow -- they're significantly slower compared to the coolant flow.

The reactor core is surrounded by a graphite reflector. The graphite reflector works as

-- to increase the neutron economy, provides moderation and reflection for neutrons. It shields the reactor structure and maintains the core geometry.

Core design methodology is described in core design and analysis methodology, the Technical Report 017.

This is some of the properties of the core. The power level of the reactor is 35 megawatt thermal. The core engines of analysis for nuclear design is SERPENT-2 continuous energy Monte Carlo code and STAR-CCM for both disparate element modeling of pebble dynamics and thermo hydraulics.

The coolant is Flibe, enriched in lithium-7. The shutdown margin is set at .99. I'll talk about that a little bit later. As pointed out by Dr. Doron, there are seven total RCSS, three shutdown elements and four control elements.

The irradiation limit -- the irradiation of the vessel is below .1 dpa, which is set in our metallic topical report as low level radiation limit. Inlet temperature is 550 degrees C. The maximum reactor outlet temperature is 650 C.

The core volume is 2 meter cubed. Enrichment of the fuel, as pointed out, 20 weight percent uranium 235. The core is a slightly under-moderated, which means negative reactivity coefficient. And also negative reactivity coefficient for the coolant temperature and void.

This seems like a busy slide, though I'll try to walk through it. There are three boxes here. One is a green box, an orange box, and a purple box. The green box is our safety tools, meaning the output of core design calculations to safety analysis is derived from the blue box.

On the blue box, there are two main engines, STAR-CCM plus, which provides the pebble dynamics through the core, and also the temperature distributions. Using porous media application and SERPENT-2 is a continuous energy Monte Carlo tool, which does neutron and gamma transport and provides the core physics parameters.

There are two internally developed methods, or I like to point, refer to them as wrappers. They basically, the KPATH is the core thermo hydraulics, which couples SERPENT to STAR-CCM to provide temperature distribution. And the feedback of temperature on power and vice versa.

KPACS feeds in pebble flow dynamics from disparate element modeling of STAR-CCM and simulates the pseudo steady state operation of a pebble bed reactor. It's very similar to how VSOP was modeled for pebble bed reactors prior to us. The difference is that KPACS works on the higher fidelity domains of both SERPENT and STAR-CCM.

The purple box is our nuclear data processing. We do process ace libraries internally. There is a verification process for them. And we do use different libraries to make -- to understand the biases of different libraries that are out there, including JEFF and ENDF 7, 1, and 8.

The support tools, the orange box is mostly used for design purposes. The KP-AGREE is spatial kinetics with thermo hydraulics feedback. It is used for understanding the behavior of the reactor as we move through different phases of operation.

And the MCNP is used for radiation transport and also for code-to-code benchmark of SERPENT. There's a tool missing on this box, and that's the scale. The scale is used internally as for code-to-code benchmark and a large space of uncertainty analysis for other neutronics domain is performed using SKID.

The core life cycle of a pebble bed reactor, it is very common across pebble bed reactors, there is -- there are four distinct phases or three --

there is an approach to criticality that we are using, a very safe approach to criticality for Hermes reactor, which we go to criticality.

Once the tests are performed, we start ascending to power. There is a phase in ascension to power which is called low power. We won't go through low power core compositions. Set of low power testing will happen. I will talk about it a little bit more on the next slide.

Once everything is done, then the reactor will start ascending to 100 percent power. Throughout those power ascension, there are number of other bullet points, which different tests will be performed. Not just core physics outside of that, also test of environment and radiation will be performed.

Once reaching 100 percent power, the reactor will -- they will stay there. And if Hermes operates long enough, ultimately the core will reach an equilibrium state, which largely the radionuclide inventory is going to stay unchanged in equilibrium.

This is the summary of what was presented in the last -- last slide. For approach to criticality, a combination of fresh fuel, natural uranium, and moderate -- moderated pebbles are added to core using a one over M approach.

Natural uranium is used as a knob for managing the effective enrichment of the core and keeping the core in a desired moderation level with respect to reactivity coefficients.

Once reactor achieves the criticality, the core composition will change to a low power core composition. The primary salt pump will follow the power. And the reactor will start ascending through different set points to 200 percent power.

Power defects, xenon, and burnup is compensated by control rods and fresh fuel addition. And as pointed out earlier, once reactor achieves -- goes to 100 percent power and if it operates long enough at that state, the core will ultimately evolve into an equilibrium core.

And there is a steady state of removal of this charged pebble, which pebbles that are reached their design burnup, and fresh pebbles will replace them at the core. All --

CHAIR PETTI: Just a question on the natural uranium pebbles.

MR. SATVAT: Yes.

CHAIR PETTI: You said you did it -- it's for reactivity control? Because you know, gas reactor pebble bits don't start up with natural uranium, they just start up with graphite pebbles and the regular fuel.

MR. SATVAT: Yes, Dr. Petti, this is another -- natural uranium pebbles are used -- if we only rely on graphite pebbles and fuel pebbles in our reactor, and fill the core with fresh fuel and graphite pebbles, if you would like to keep excess reactivity to a desired level, the core will be over-moderated.

So we are using natural uranium pebble as a mean to keep the effective enrichment and carbon to -- carbon to heavy metal alloy ratio in a desired place.

CHAIR PETTI: Okay, I got you.

MEMBER KIRCHNER: Yeah, Dave, I was going to ask the same question. So these natural -- natural uranium pebbles, what's the fuel form?

MR. SATVAT: It is -- they're exactly the form of a TRISO particle as it is in our enriched fuel.

MEMBER KIRCHNER: So you're going to actually have a TRISO line that uses natural uranium?

MR. SATVAT: Yes.

MEMBER KIRCHNER: Okay. And this will

prevent an over-moderated state?

MR. SATVAT: Precisely.

MEMBER KIRCHNER: Why couldn't -- why couldn't you -- never mind, okay, thank you.

MR. SATVAT: To add to that, the HTRPM, the Chinese reactor, they're actually using different enrichment. They're using lower enriched all the way up to their highest enriched for a startup, and they continuously remove lower enrichment.

That might be something we consider for future larger reactors, but for Hermes, we found this to be a desired approach.

MEMBER KIRCHNER: Yeah, that was done in other reactors to change the enrichment level to avoid the over-moderation problem. Okay, thank you.

CHAIR PETTI: And then just a question, you said you'd check the reactivity coefficients on startup. But is there any plan, or is it easy to do it as you approach equilibrium?

MR. SATVAT: The isothermal temperature reactivity coefficient will be tested during low power regime. Currently the detail of all the physics testing that will be performed is being completed and will be provided as part of operating license application.

But we believe if we perform the isothermal throughout the low power, we can rely on our predicted models not to have to continuously measure it as we move through different core compositions.

CHAIR PETTI: Okay.

DR. SCHULTZ: Nader, this is Steve Schultz. Can you give an idea what the timberline is for moving through these various stages approaching criticality for Hermes, going through the ascension to power, and the -- how long does it take to get to the equilibrium core, as you see?

MR. SATVAT: That's a very good question. So the time that it takes to get to equilibrium is basically the -- where the core will assume an average burnup distribution --

DR. SCHULTZ: Yes.

MR. SATVAT: Of the equilibrium core. It is -- in our calculations, usually it reaches around 1.5 times the residence time of the pebbles. So if we ascend to power rapidly and stay at 100 percent power, whatever the residence time is is going to be close to 1.5 times that is the time that it takes to get to equilibrium.

DR. SCHULTZ: And that, you mentioned that could be around 40, 50 days, something like that for the residence time?

MR. SATVAT: Thirty, 40 per pass, which means we're designing to six pass. And again, those details will be refined for oil. But as of now, there are six passes. So if there are 50 days, there are about 300 days of residence time.

DR. SCHULTZ: Okay. Okay, thanks, that helps a lot.

MR. SATVAT: Sure. As a last point on this slide, all core states will operate within coolant temperature to coolant reactivity coefficients power for particle limits and excess reactivity constraints.

The design basis. There are two important PDCs that are met in this slide. The reactor core is designed so that the power oscillations that could result in conditions exceeding SARRDLs are not possible.

This is effectively due to the small size of the core of Hermes and the very long neutron migration length, which means the core is effectively neutronically connected. There is no credible way of having oscillations that -- that are seen in large LWRs.

The reactor core is designed so that the net effect of prompt in paranuclear feedback tends to compensate for rapid increases in reactivity.

This is PDC-11. The overall reactivity coefficient is negative. It is provided -- the list is provided for fresh core and equilibrium core at the -- at the table below. The fuel Doppler is negative. The moderator is negative, so is coolant and the void.

The reflector temperature coefficient is positive. The positive reflector temperature coefficient is due to a spectrum hardening shift, which shifts flux towards the center of the core. It is due to reduce leakage. Also because of the locally over-moderated conditions by the reflector, periphery of the core.

Methodology does not assume any thermal expansion of reflectors, so it's actually conservative. It is on the higher end of being positive, the reflector temperature.

The reactivity impact due to the reflector temperature is delayed compared to fuel and coolant temperature feedback.

But at the last point here again, the overall feedback coefficient respond to temperature increase is negative at all conditions in -- in Hermes reactor.

The next PDC, PDC 10, a limiting power distribution for the core design is used to ensure that the reactor has appropriate margin to SARRDLs. As pointed out in KP-FHR core design methodology, SERPENT-2 is used to calculate power distribution. Flux distributions are verified during startup and low power using the ex-core detectors.

Flux measurements compared to predicted values will ensure core is operating as designed. There are no consequence from control rod -- control and shutdown elements not being core, they're core symmetric.

And I'm putting that picture down there. As you can see, the yellow RC is the four of them around the core. They're not fully symmetric, but that's not causing an issue in a small core such as Hermes with long neutron diffusion length.

There are some example calculations provided in PSAR for -- in (audio interference) pebble peaking factor.

This is a busy slide, but bear with me. It's PDC-26. That first condition, shutdown elements credited to provide means to ensure SARRDLs are not exceeded and safe shutdown is achieved. This is assuming highest worth shutdown element fully withdrawn. So with only two shutdown element, we achieve this.

Shutdown elements insert reactivity at a sufficient rate, an amount to ensure the capability to cool the core is there. The reactor is shut down and can be maintained in a shutdown condition. It is again met with the highest worth element fully with the rod or stuck.

The second condition, control elements provide the capability to control reactivity change during normal power changes. This ensures SARRDLs are not exceeded and provide an adequate and separate means of reactivity control from RSS.

Control elements are diverse from shutdown elements. They have different geometry, different location, different insertion mechanism.

Condition number three, shutdown elements insert reactivity at a sufficient rate and amount to ensure the capability to cool the core is maintained. The reactor is shut down and can be maintained in a shutdown condition. This is again done with the fully -- with the highest worth element stuck out.

Shutdown elements provide a means of maintaining the reactor in a shutdown state to allow for fuel loading, inspection, and repair.

And this is basically continuation of previous slide. The shutdown margin calculation accounts for power defect. The decay of xenon, operational excess reactivity and margin for uncertainties are incorporated in the calculations.

As pointed out before, the shutdown core is defined as k-effective of .99. The required worth for shutdown of the system is 11578 pcm. The total worth of shutdown elements is 14232 pcm.

And again, this is -- this is the required worth considers highest worth element with the rod. And as it can be seen, it's the first of a kind reactor, we recognize that the relying on general purpose nuclear library might introduce uncertainties. So we do have the required shutdown is 1000 but we do have 3652 of margin for shutdown.

The interfaces, these are the nuclear design, nuclear analysis interfaces that the -- is connected to the methodology. The vessel fluence supports reactor vessel design. Fluence on vessel accounts for core pebble insertion. Extraction lines' fluence is attenuated -- attenuated by the core barrel, reflector, and the coolant.

Preliminary best estimate DPA plus

uncertainty is shown to be 30 percent below the low level irradiation value provided in metallic material qualification for Kairos power.

Nuclear transient analysis, which supports the safety analysis conservative values used for power distribution, reactivity coefficient, and shutdown margin is provided as initial conditions for postulated reactivity transient events.

In -- they're provided in Table 7-1 with uncertainties and nuclear reliability factors as an output of the methodology.

The core design limits, which supports technical specifications, core design parameters during normal operation are shown to be within fuel qualification envelope for peak fluence, peak particle power burnup, and peak fuel temperature. Shutdown margin. Coolant outlet temperature moderated pebble to fuel pebble ratio.

I think that's it.

CHAIR PETTI: Members, any questions?

MEMBER KIRCHNER: Just one, Dave, from me again. How do you know where the moderator pebbles are, or now the uranium pebbles? In the HTGRs like THGR, the pebbles were dropped in kind of just randomly kind of built a packed bed core.

Here, you're going to inject them from the bottom, I believe. Is the bed packing so tight that you have a good experimental basis for determining where the pebbles actually distribute themselves?

MR. SATVAT: That's the topic of disparate element modeling that we have for pebble dynamics in the core. We do have set up -- set up experiments internally to validate other models. So far, the packing fractions for escape tests that we have performed showed -- showed to give us a packed bed, close 60 percent.

MEMBER KIRCHNER: Yeah.

MR. SATVAT: So to answer that question, yes.

MEMBER KIRCHNER: Okay. Okay, thank you.

CHAIR PETTI: Just one more question. Your slide on the codes you're using, maybe it's because it's just a design. Didn't mention SAM, but the document did. Is that -- that's a -- I don't know, system-level code I guess on safety analysis.

MR. SATVAT: That's correct. These are steady state calculations that are -- that the boundary conditions are transient analysis. You are correct, Dr. Petti, those are done in our transient methodology, KP -- using KP-SAM.

CHAIR PETTI: Okay, got it, thanks. Any other questions, members?

Okay, then with that, staff.

MR. BIELEN: Hello, good afternoon, everyone. My name is Andy Bielen. I'm a Senior Neutronics Engineer in the Office of Nuclear Regulatory Research. I'm going to be presenting here with Ben Adams a review of PSAR Section 5.4, nuclear design. So, next slide, please.

So basically what we're going to do, I'm going to first talk about the review of the methodology technical report that they provided, which you know, a substantial fraction which is proprietary, so we'll try to keep this as a high level discussion. Then I'm going to go through some coping -- scoping analysis that the staff performed with the scale code suite.

And then I'm going to turn it over to Ben so he can talk about the PSAR content and our findings thereof.

Okay, so the regulatory basis is very similar to the other sections, or the other sections in Chapter Four, so the look through the preliminary safety analysis report. We have 50.35 that issued the construction permit, etc., etc.

We have our non-power reactor SRP and then for PDC from Kairos's PDC topical report.

Okay, speaking of topical reports, so the PDC topical report is number one. Then there's a material, or metallic materials qualification topical report that's specifically applicable to the fluence limits. And then also this technical report, which is not a topical report, but is tied to the PSAR itself.

Okay, so I think -- I think that Kairos did a really good job of, you know, describing their methods. I don't want to get too far into the weeds on like what they're doing, other than to say within the methodology, we have two remaining physics codes, SERPENT-2, which is the transport code, and isotopics, it's based on the continuous energy Monte Carlo method.

They used Doppler broad and cross-section data provided up front by their ace library generator code. Then they'll also use STAR-CCM plus for doing the DEM solution of -- or DEM predictions pebbles as they move through the bed. And then also a porous media approach to calculate temperature and coolant distributions, or coolant density distributions.

And as -- as they said, there's two analysis sequences. There's KPATH, which is iterations between SERPENT and STAR-CCM for comparable neutronics and thermal fluidics.

And then they have KPACS, which is what they use to do their fuel cycle analysis. So they evolved the isotopics within the geometry using -- using KPACS based on what the DEM tells them about how the pebbles move throughout the core.

Okay, so in terms of our evaluation of this methodology, I think the, you know, big picture, we think that it's a sound methodology. We think that they've accounted for the things they need to account for. I think we're not overly -- they're calculating a lot of different cross-section libraries based on temperatures, so they're not overly relying on on-the-fly interpolation.

They have adequate treatment of axial and radial zoning so they can capture spectral effects appropriately. Also accounting for the isotopics within different passes, depending on how many times they've gone through the core.

The coupling to the thermal fluidics is -- seems to be reasonable and acceptable. We think that the models as a whole provide a robust means to calculate the things that they need to provide a transient analysis and also provide their shutdown margin and reactivity coefficient, etc., etc.

And you know, one -- one point I would say is that, you know, when they calculate some of these feedback coefficients, they're doing it in a conservative way.

For example, you know, the specific composition of the Flibe as they're calculating the coolant temperature feedback would result in a -- in a conservative evaluation of that specific feedback coefficient. So yeah, I think the big picture, we think they're on the right track.

There are some, you know, remaining methodology items that I think that we need to close before we get to OL stage. As I've mentioned, or maybe didn't mention explicitly, but a lot of the PSAR analysis is -- is focused on kind of a nominal average, if you will, core performance. But as you know, these are stochastic machines.

So there's a -- some kind of uncertainty elements that need to be accounted for, both in the underlying physics, but also in, you know, the way that it would be operated in the different limiting, you know, configurations of particles that might randomly spawn themselves as they're -- as they're reloading this thing, you know, from the bottom.

So one element that they have to do some more work here is, you know, establishing the neutronics uncertainties. You know, as a starting point they've got a list of what they feel based on engineering judgment are conservative uncertainties and their key, you know, neutronics figures of merit, which feed into the safety analysis.

So you know, they need to -- they have additional work planned to verify the conservatism in those, you know, the assumed uncertainty in those different parameters. We expect to see, you know, some more details of that work at the OLA stage.

And another big, you know, missing item from the staff's perspective, and I know they've done some work internally on this thus far, but we have not yet seen anything specific on assessment or validation. Though there's descriptions of the validation plans within the -- within the technical report. And we want to -- we want to make sure that we get a chance to see that before we get to OL stage.

And then there are some kind of nitty, you know, during the review and during the, you know, the audit period, there were some of kind of nitpicky things that we were, you know, asking them questions about on specifics of various modeling approaches and just verifying that those modeling approaches were robust.

So you know, I think that those are things that they're on -- they'll be able to close, they're on track to close. But just kind of some things that we want to make sure that are accounted for before we get to -- get to the OL.

So you know, as such, I think we just want to be clear that right now, Kairos's neutronic design approach is not, quote unquote, reviewed and approved. But given closing these items, we think that they're well on their way to get there I guess, so.

So before I get into the scope and calculations, do you have any questions that I can just get or answer?

MEMBER HALNON: Just real quick, this is Greg. This alphabet soup of codes and stuff, it all seemed reasonable when you played it out? And did you guys diagram it out as well and make sure that there's no big hole in there from the standpoint of one code's beating another and that sort?

MR. BIELEN: Yeah, I think that like the way that the codes communicate within each other needs to be part of that validation and verification assessment --

MEMBER HALNON: That's what you'll be looking -- and I figure that we're not sending goofy data to one place and coming up with a good result and saying everything's good.

MR. BIELEN: That's right. I mean, and you know, when it comes to coupling codes or sending -- doing data exchanges between codes, when you make a mistake, it's usually obvious. But it's -- that's not guaranteed, obviously. So you have to -- you have to go through your QA process and make sure that you're doing that appropriately.

MEMBER HALNON: Excellent, thank you.

MR. BIELEN: Okay. Well, I'm just going to walk through real quick here some scoping calculations that we were able to do to support the CP review.

So hopefully the members are remembering that over the last several years, the Office of Research and the agency in general have been, you know, kind of tasked with getting our codes and methods analysis procedures prepared for non-LWR licensing.

So since 2018, 2019, 2020, we have, you know, several different areas of applicability and then different codes within those areas of applicability that we need to make sure that we can do non-LWR confirmatory and licensing support analysis with.

So part of this non-LWR vision and strategy, we have several different volumes. Volume 3 specifically covers severe accidents in source term, which -- of which we have several demonstration plants depending on different technologies.

We happened to perform analysis on the publicly available UC Berkeley Mark 1 design, we represented a Kairos-like pebble -- TRISO pebbles with molten salt cool -- coolant. We used the SCALE code suite for decay heat and radioisotopic inventory and then reactor physics data performed by Oak Ridge National Laboratory.

And then our MELCOR code was used to simulate -- different accident progressions using the data that SCALE provided for it. And that was performed at Sandia.

So we did a demonstration workshop about a year and a half ago fully documenting this specific application.

And the good news is because we did that work ahead of time, we were able to rapidly, at NRR's request, adjust our models to be more Hermes-specific and be able to perform an analysis like fairly quickly that I think really helped them form their engineering judgment and feel more comfortable and confident in the -- in the results they were seeing from the applicant. So, go to the next slide, please.

So one thing I want to kind of cover here, cover off the bat is this is, we're calling this scoping rather than confirmatory. First of all, given the licensing timelines we have for this reactor, we wanted to be able to do this within a rapid turnaround period.

So in fact, the initial discussion started in December of 2021. We started doing work in January of 2022. And the first briefing to NRR staff, you know, was by the end of -- the end of March of 2022. So we were able to do this within three months, which I think is, you know, fairly notable.

We want to, you know, we wanted to basically not go through the RAI process in case there were data that we -- data that we needed to have. So basically we did the best we could with what was available publicly in the Hermes PSAR.

Plus, you know, wherever there were gaps, you know, engineering judgment or applicable data from the -- from the UC Berkeley designed, you know, I think was able to come up with at least a reasonable surrogate for this model without -- while we all understand that, yes, there's going to be some differences. And those are probably reasons for some of those differences.

Now, I'll just point you to this -- this report from Oak Ridge on the overall SCALE FHR workflow. It's very similar to our work flow for the pebble bed gas reactors.

But basically, we're justifying a multi-group rather than continuous energy treatment. Then we described the process of generating an equilibrium isotopic inventory using 2D models that we then can feed into a full 3D core model and then do reactor physics stuff with that.

And you know, before I go too much further, I just want to make sure that I acknowledge Rike Bostelmann at Oak Ridge National Laboratory. She really led this work there. And you know, provided most of this analysis and explanation, and hopefully I don't butcher what she did too badly.

But the point is that we have this model of the FHR system that is a -- as close to Hermes as we can get based on what is out there in the public. We're using multi-group Monte Carlo transport using

KINO-6 for isotopics and you know, evaluating as how the isotopics evolve with operation.

So in the real system obviously it's a random packed bed. You know, given our current capabilities, like we have to make some assumptions and approximations in order to have a tenable run time for our model. So we're approximating these with a regular lattice.

As I mentioned, we use this 2D slice model with SCALE/TRITON to produce the TRITON sequence out of the SCALE code to produce equilibrium of isotopics. So these are -- so what you're seeing in this model is basically the isotopics vary axially from bottom to top.

Like each level of pebbles, even though, you know, we know that their -- different passes of pebbles will have different isotopics depending on what their specific history is. We're kind of smearing all that out into, you know, an axial average isotopics at each level basically.

So you know, it's nearly as high fidelity as I would say the SERPENT model that the applicant is using is. But you know, big picture, I think that it's adequate to get us the sort of information that we're interested in seeing.

And another, you know, item of note here, we don't currently have the shutdown in bed elements within this model, which is obviously something that we are working on, but we haven't gotten that complete for -- in time for this presentation.

But you do see the -- we have those channels outside the core that would represent where the control elements go.

MEMBER HALNON: I'm getting confused a little bit, but weren't there three types of pebbles now? Fuel pebble, the natural uranium pebble, and a moderator pebble? Is there any distinction there?

MR. BIELEN: So I think the natural uranium pebbles were only in the startup core. So we're basically, we're performing an -- like the way that we generate these is to go directly to equilibrium.

MEMBER HALNON: Okay.

MR. BIELEN: So yeah. That's why we wouldn't -- we could do that if we wanted to. But I think, you know, from a limiting, you know, state point before an accident scenario from a decay heat, you know, radio inventory, radioisotopics point of view, the equilibrium is like a limiting --

(Simultaneous speaking.)

MEMBER HALNON: And that's the equilibrium that we asked earlier about, which was 300, potentially 300 days?

MR. BIELEN: Something like that.

MEMBER HALNON: So in the interim, you say everything is bounded by that 300-day equilibrium?

MR. BIELEN: I think that that's true in general. There may be some specific things that aren't, but you know, I --

CHAIR PETTI: Inventory-wise, I agree with you. But there may be events during startup that challenge the systems in unique and different ways than they would during --

(Simultaneous speaking.)

MEMBER HALNON: That's what I was thinking about too, especially with the unknowns.

CHAIR PETTI: Well, and you know, the decay heat removal system has a certain capacity, right. And then you're generating energy. At steady state, everything looks good.

But now, you know, you're at 7 megawatts thermal or 10 megawatts, you know. Can you make sure you don't remove too much heat to freeze? Those are the sorts of things that I think are more interesting to look at analytically.

MEMBER HALNON: That's noted, we can try to adjust that going forward, so. Okay, so you know, just to kind of walk through some of our results here --

MEMBER BROWN: This is Charlie Brown, can I ask you a question?

MR. BIELEN: Sure.

MEMBER BROWN: I don't know whether -- I don't know whether Dave and Walt and the others, I'm not a fuels guy. But how in the world in this thing with three or four or whatever it is types of pebbles do you get any types of predictability on a uniform distribution with these things moving up through the core?

You said you'd get an equilibrium, but I, it's just hard to envision all these different pebbles being fed in and then achieving any type of uniform distribution throughout the core region. How do you deal with that?

MR. BIELEN: Yeah, I mean, so the way that Kairos specifically deals with it I think we're going to have to, you know, I'm going to defer to them to discuss with you whatever proprietary details that they want to share in this environment.

But I will say that in general, you know, you have so many pebbles that it does become a very -- it's a very kind of law of large numbers statistical process I guess.

Like, and that's the best way I can, you know, kind of describe it off the top of my head. Like I think that -- I think that in general you can represent the average behavior of the core and -- and the pebbles, you know, using -- using kind of averaging or stochastic methods.

And then for those special cases where hey, well, what happens if like there's some bunch of fresh pebbles get by chance randomly, you know, loaded next to each other or something like that.

In that case, then you would want to do like a special calculation or like a detailed calculation of that specific scenario to prove to yourself that you're, you know, meeting your thermal margins.

CHAIR PETTI: So Charlie, this is a question that's been around for a while. And a tremendous amount of work has been --

MEMBER BROWN: That's why I asked.

CHAIR PETTI: Has been done in the gas reactor realm. The concern at the time was all about the fuel, okay. They didn't know what the peak temperature limits were for some of the fuel. And UO2 TRISO had more restrictive limits.

And so knowing all these distributions well was more important. My personal view is with UCO TRISO, the temperature, a lot of these questions go away because, okay, we may -- you don't think you know it. But there have been tons of studies, tons of codes, international benchmarks on all of this stuff to show you do get to this, quote, equilibrium.

And what that means is just on average, okay, on average at a location in the core, that condition stays at that condition all the time. If there are seven pebbles in that volume, they can had different burnups. But on average you get the same -- the same number.

MEMBER BROWN: So you don't have to -- you don't have to worry about a non-equilibrium temperature distribution that's too far out of whack.

CHAIR PETTI: Correct. If your fuel had really tight limits and you were close to fuel limits, then it's a big issue. But I really think that a lot of those types of questions don't matter with the TRISO fuel today. Which is really good, because it adds a lot of questions.

MEMBER KIRCHNER: Yeah, Dave, you're

right, you've got a lot of margin. The other thing is that there's so many pebbles, Charlie, that it's, you know, it's almost like a random number generator.

And then the other thing that was pointed out earlier is that the core's, at least for this size and for this power, is very well coupled. So that the local spatial -- the local heterogeneity of the pebbles isn't as critical as when you start pushing a system in a larger core configuration.

MEMBER BROWN: Is there a larger core --

MEMBER KIRCHNER: Because the neutron diffusion life is quite long. So it's a well-coupled core, even though it's heterogeneic pebbles.

MEMBER BROWN: Okay, all right. I'm obviously not fuels, it's just, I just had to ask that question at some point as we went through this, that's all. I've listened, I will be quiet.

MR. BIELEN: Yeah, so I think just kind of going over some of the results in comparison with what's in the PSAR. So I think we do very well with the axial power distribution. The radial and peak pebbles, there are some differences.

And I can tell you that I think the radial difference is due to a difference in the way that Oak Ridge defined the radial peaking factor versus the way that Kairos did. But unfortunately Kairos's definition was behind the proprietary wall. So you know, if that ever gets lifted then we can -- we can update that accordingly.

And then, you know, there's also, you know, we don't know exactly what the peak, or the radial reflector looks like. We don't know where the, you know, how much Flibe is in the reflector.

It may change the neutronic characteristics on the edge of the core where the peaking really is. So, you know, that may feed into, you know, the difference in peak pebble peaking.

But you know, given all the uncertainties we have in this and the, you know, the limitations and constraints we were working under, we thought these were, you know, reasonable comparisons that could be, you know, further understood or resolved, given additional information.

Okay, and then, you know, the big thing for transient analysis is reactivity -- reactivity feedback coefficients. And I think that, you know, this slide shows despite all these differences in modeling and in energy treatment and in geometry and isotopics, etc., etc., we are still like getting remarkably good agreement in most of our transient coefficients.

So you know, I think that that kind of gives us a lot of comfort that, you know, the analysis that we're seeing from the Applicant is at least on some level, you know, consistent with other sources of information. So I think that that really helps -- helps us, you know, justify and defend our acceptance of their -- of what they're presenting, so.

MEMBER HALNON: Did you do any sensitivity analysis on that to see how if changing might change your results to where they become not remarkable anymore?

MR. BIELEN: I would say most of our sensitivity analysis -- so this, full disclosure, this isn't the complete set of comparisons that we formed. We, you know, are showing the ones that we think are most important for this context. But like there are certain areas where we have results that aren't as good as we would like.

And I think we spent most of our, you know, whatever limited budget we had left to do sensitivity analysis after this was done, we spent more time kind of focusing on those things. Like you know, okay, is it important to capture, you know, the difference in differential speed between pebbles going up the middle versus pebbles going up, you know, adjacent to the -- to the reflector.

And you know, okay, well, what about the composition in the reflector for the pebble peaking on the periphery and all those sorts of things. So I think that that's a great point.

You know, this is, okay, you could accuse us of confirmation bias to some degree because we saw what we liked and we said oh, okay, this is great. But you know, I think this wasn't all that we saw and some of what we saw we didn't like. So, you know, we did explore some of those things a lot deeper.

MEMBER HALNON: Thanks.

DR. SCHULTZ: Andrew, the -- this is Steve Schultz -- the uncertainties you showed is noted as the Monte Carlo uncertainty. Other bases of uncertainty that you would explore in order to determine the goodness of the comparison that you show here?

MR. BIELEN: Yeah, I mean, we would have to basically go through and do like a nuclear data based on certainty or manufacturing, you know, tolerances based on certainty. And really, like we have tools in SCALE to do that stuff but we just, you know, haven't gotten to that point yet.

So given the degree by which, you know, Rike was able to converge these, like the neutronics solutions within these models and then propagate that uncertainty into -- into the uncertainties in reactivity coefficients, like that was pretty much where we were able to end.

So and we expect during, you know, Kairos's assessment of their uncertainties in their parameters, that that would be something that they would -- that they would be looking at, you know.

DR. SCHULTZ: Sure.

MR. BIELEN: Physics-based uncertainties on these -- on these parameters.

DR. SCHULTZ: Well, this gives a real good feel for how the models work and you've done it independently, and that's excellent demonstration. And it's good to present it in public forum, that's excellent.

CHAIR PETTI: And that you know, for other members, there are benchmarks in the IAEA on pebble bed reactors. So there's a lot of codes out there, a lot of countries that participated. You know, it's a lot different than, say, 20 years ago.

MR. BIELEN: Ben, your turn, buddy.

MR. ADAMS: This is Ben Adams again.

The way the PSAR and SE are laid out is it goes through topics like power distribution, the shutdown margins. So we're going to start with power distribution and go from there.

The codes, like we talked about earlier, SERPENT-2 is used to calculate the core power distribution and STAR-CCM+ is used to calculate the material temperature distributions, and those two codes are coupled together.

The core power distribution is characterized by the axial peaking factor, the radial peaking factor, and the total peaking factor. Those are calculated using SERPENT-2. We just show a bit of our comparison for that.

That scoping calculation showed reasonable agreement with Kairos's preliminary calculations of power distribution and total pebble peaking factors. The core power distributions are an input to the fuel performance calculations, and the staff finds that this is consistent with PDC-10.

Okay, the shutdown margin criteria is defined as k-effective less than 0.99 as defined relative to the margin to Flibe freezing.

The control element worth is calculated from changes to k-effective resulting from determining the element axial positions to SERPENT-2 with the single most reactive element assumed to be fully withdrawn from the core. It's a pretty standard method from what we can talk about in the public forum.

Kairos will perform source range control element worth testing. And the staff finds that this is consistent with PDC-26.

The reactivity coefficients again were calculated using SERPENT-2. I won't list them off again. They are all negative during startup and equilibrium, except for the reflector coefficient, which is always positive.

The reflector coefficient is only slightly positive, but like I believe Kairos mentioned, thermal expansion is conservatively forward because that would be a difference in leakage in that the moderator temperature reactivity coefficient includes the change in reactivity due to the change in temperature in the fuel pebble graphite and in the graphite pebbles.

And the coolant temperature reactivity coefficient accounts for density changes. And the coolant void reactivity coefficient is the change in reactivity due to the change in coolant void fraction.

Overall, reactivity coefficients are

negative even though the one reflector coefficient is positive always. The staff looked at scoping calculations to show reasonable agreement with Kairos's preliminary calculations for the reactivity coefficients, which we also just looked at.

Once Hermes achieves criticality and is at zero power, Kairos will perform isothermal remote reactivity coefficient testing to confirm those. And staff finds this is consistent with PDC-11.

For the OL, we will be looking for how five compositions impact the reactivity coefficients as well as the graphite densities in the reflector. Our scoping calculations do not account for different Flibe compositions from what I recall.

And we did perform some sensitivity studies on the graphite density and it did have an impact. So those are a couple of things that the staff is interested in looking at during the OL stage.

The vessel lifetime was originally slated to be ten years but was reduced to four years during the course of the review. The vessel is showed by the core barrel, the reflector, and the Flibe coolant. SERPENT-2 is used to calculate the fast neutron fluence and alpha generation on the vessel received from the core and pebble insertion and extraction lines.

And Kairos's preliminary calculations of displacement per atoms in the vessel is within the low level irradiation value discussed in KPTR-013. Kairos did not give us the full methodology for calculating vessel fluence, including how uncertainties are accounted for. So we will be looking through at the OL stage.

We did not do any scoping calculations for this. And we'll review this in greater detail later at the OL stage.

The nuclear transient perimeters are outputs from SERPENT-2 and are used as inputs for transient analyses. They include the neutron generation time, the delayed neutron fraction groups, and their decay constants and the prompt neutron lifetimes.

They were given the PSAR for both equilibrium and startup. These perimeters play into things like the kinetic responses and the timings of transients and the shapes of curves related to the kinetics model.

Our scoping calculations did have relevant agreement with Kairos's preliminary calculations. So we were comfortable with this level of information that was given in the PSAR. And these scoping calculations, again, will be discussed with Chapter 13 later on April 18.

Kairos notably did not submit an uncertainty analysis for nuclear transient parameters. We will be reviewing that later as well.

The staff's evaluation of the nuclear core design limits of the burnup peak fuel temperature, peak particle power, and peak fluence are discussed in Section 4.2.1. We talked a little bit about that earlier today.

The neutron flux will be monitored using four power range ex-core detectors located in azimuthally symmetric locations outside the reactor core at mid-core elevation. And four source range ex-core detectors located in relation to the start of source for best detectability criticality.

During normal operation, these detectors will be used to monitor core power and the flux rate trip signal. And the source range detectors will be used during startup.

Gamma spectroscopy will be used to evaluate the fuel pebble burnup. But we did not make any findings on the burnup monitoring plan in this section. I believe that is discussed in Chapter 9.

Kairos will also perform neutron flux distribution verification tests during startup. And staff finds that this is consistent with PC-10.

For the technical conclusions, we discussed some of the PC. Preliminary design information provided is consistent with PDC-10, 11, 12, and 26. And is consistent with some of the acceptance criteria in NUREG-1537. And the analytical methods for the nuclear design are acceptable at this stage of the design.

And for the regulatory findings in Section 4.5, NRC staff concludes that the preliminary design provided in PSAR Section 4.5 is consistent with the applicable PDCs and acceptance criteria in NUREG-1537.

The NRC staff finds that the information in Hermes' PSAR Section 4.5 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40. And further information can be reasonably left for the OL application.

I believe that is the last slide.

DR. SCHULTZ: Ben, the statement that you didn't make any findings related to the burnup monitoring plan that Kairos has, is that because more details are required and they will be available at the OL stage?

MR. ADAMS: Yes. Part of it is because I believe we addressed it in Chapter 9 and not in this section. And also because the details are -- they're not very detailed, so we will be reviewing that in detail at the OL stage, yes.

DR. SCHULTZ: More information needed.

MR. ADAMS: Correct, yes.

DR. SCHULTZ: I appreciate that, thank you.

MR. ADAMS: Thanks.

CHAIR PETTI: Members, any other questions?

Okay, with that, I think we're done for the day, 40 minutes early is good. And -- huh? Oh, public comments, yeah, we probably should. It's a good idea.

Okay, so anybody out there, a member of the public that wants to make a comment, please unmute yourself, your name and your comment. That's coming from Kairos. Yeah, I'll get him.

Okay, not hearing anything from the public. Kairos?

MR. TOMKINS: So can I -- this is Jim Tomkins, Manager of Licensing. This morning we had a discussion about the burnup in Hermes. And we mentioned that it's around 6 percent FIMA.

There were some questions about what that corresponded to as far as operating plants. So Nader is the expert we didn't have then. So if you could -- you could talk on that.

MR. SATVAT: For Hermes, every percent FIMA is equivalent to 9.4 gigawatt day per metric ton. So for Hermes, it's about 50-60 gigawatt day per metric ton.

Our power plant, the full scale -- full scale plant will go up to 20 percent FIMA, close to 20 percent. So it's about 180-200 gigawatt day per metric ton.

CHAIR PETTI: Thank you.

MR. TOMKINS: Sure.

CHAIR PETTI: And with that then I guess we're done for the day and we'll see everyone again same time, same place tomorrow morning. Thank you, everyone.

MEMBER KIRCHNER: Dave, just quickly, could you just go over the chapters that are going to be covered tomorrow?

CHAIR PETTI: Sure. So tomorrow we're going to finish the rest of Chapter 4, so Section 4.3, 4.4, 4.6, and 4.7, as we did 4.2 and 4.5.

Then all of Chapter -- that's before, all of them before lunch. Our memo will be the last thing before lunch. Then after lunch, we'll do Chapter 6 completely. And with our memo, break, then Chapter 9. Then public comments, and then we still have a closed session, but I doubt that we'll need it. But it's there.

MEMBER KIRCHNER: Thank you.

CHAIR PETTI: Okay.

MEMBER DIMITRIJEVIC: It is Chapter 6 completely, Dave, or it's just 6.2 and 6.3

CHAIR PETTI: Proprietary? The whole thing, the whole thing.

MEMBER DIMITRIJEVIC: So all Chapter 6.

CHAIR PETTI: All of Chapter 6.

MEMBER DIMITRIJEVIC: All right.

CHAIR PETTI: Okay, everyone, have a good evening. Thank you.

(Whereupon, the above-entitled matter went off the record at 4:22 p.m.)

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards Kairos Power Licensing Subcommittee
Docket Number:	(n/a)
Location:	teleconference
Date:	Friday, March 24, 2023

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

KAIROS POWER LICENSING SUBCOMMITTEE

+ + + + +

FRIDAY

MARCH 24, 2023

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The Subcommittee met via Videoconference, at 8:30 a.m. EDT, David A. Petti, Chair, presiding.

COMMITTEE MEMBERS:

DAVID A. PETTI, Chair

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

WALTER L. KIRCHNER, Member

JOY L. REMPE, Member

ACRS CONSULTANTS:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

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

8:30 a.m.

CHAIR PETTI: Okay. I guess we're going to restart. As planned, everyone, we're going to finish up Chapter 4 before lunch and then hit Chapters 6 and 9 after lunch. We should just start with Hermes, I guess, right? You don't -- okay. So, yes, I'm sorry, Kairos, you're up.

DR. DORON: Okay. This is Oded Doron again, Senior Director of Reactor System Design, and I'm going to be presenting on Section 4.3, the Reactor Vessel System.

So the reactor vessel is -- the vessel, the head, the shell, are all made out of 316H stainless steel. The vessel material is qualified for our metallics topical reports. That's KP-TR-013. The reactor vessel top head supports attachment of equipment and components. It's bolted and planned, designed to be leak-tight, but it's not credited for that. The head nozzles and attachments are seismically qualified, and the hold-down structure, that's also 316H, and it provides support against upward buoyant loads from the graphite.

CHAIR PETTI: Question on the hold-down structure. Is that kind of like a cage structure?

What's it physically look like?

DR. DORON: I don't think we went into detail on the PSAR, per se, but think of it as simple as possible. Think of it as a rib system that's just maintaining the upward buoyant loads from the graphite, and it has pins, as well, but the graphite, of the top layer of the graphite attaches to, or engages with rather.

CHAIR PETTI: Is there any concern on temperature limits there? Because that's pretty hot at the top of the graphite and stainless steel temperature limit.

DR. DORON: We don't believe that we'll be hitting temperature limits there from the work that we've done. But, certainly, I mean, it's a consideration for sure.

CHAIR PETTI: Yes. I mean, every high-temperature reactor I've seen, the metals in the hot part always can be challenging.

DR. DORON: Yes, for sure.

MEMBER BALLINGER: This is Ron Ballinger. You can be assured that, during the OL review, those temperature limits and history and the like will be of great interest, at least to me.

DR. DORON: Yes. They're of great

interest to us, as well. To add a little flavor to it, they're in the gas space. They're not in contact with the FLiBe at that level.

MEMBER BALLINGER: That doesn't bother me.

DR. DORON: Okay.

CHAIR PETTI: Yes, that actually could make it worse. I mean, I'm assuming the FLiBe will sort of even out any peaking, so it will be interesting to see what the numbers look like.

DR. DORON: Yes, yes, absolutely. I mean, it's within design considerations right now, and you will get many, many more details on it. Okay?

CHAIR PETTI: Yes.

DR. DORON: The reactor vessel shell maintains the reactor coolant boundary and provides a geometry for coolant inlet and heat transfer, the heat transfer surface for the decay heat removal system, DHRS. The reactor vessel bottom head maintains the reactor coolant boundary, provides flow geometry for the low-pressure reactor coolant inlet to the core.

I just want to add a note here, and it will make more sense when I go to the next one that the lower head, in conjunction with the shell and the core rail is what creates that inlet path. The gap that exists between the core rail and the shell and then the gap that exists between the core barrel and the lower head, if you can visualize that, that's where you have the flow. There's a figure on the right --

MEMBER KIRCHNER: Can we go -- this is Walt Kirchner. Can you go back to what you precisely mean by being designed for the top head to be leak-tight but not credited? Because your argon cover gas system, if it's not leak-tight, then that would be a concern.

DR. DORON: So we design it, and there's a seal there, it is designed to be leak-tight. We don't design it to leak, but we do not credit it as a safety-related leak-tight boundary.

MEMBER KIRCHNER: I have to think about the implications of that. That means you don't test for leak-tightness.

DR. DORON: Yes.

MEMBER BALLINGER: This is Ron Ballinger again. If it's a Section III weld or B31, regardless of whether you credit it or not, you're going to have to weld it in accordance with that procedure, and that will require a test.

MEMBER KIRCHNER: Right. That's where I'm going, Ron, yes. I mean, it's a Section III vessel.

Are you saying that the top head and flange is not Section III?

DR. DORON: No. The connection of the top head to the vessel is bolted and flanged.

MEMBER KIRCHNER: Well, yes.

DR. DORON: Everything is per Section III, but I do have a seal there. That seal is not credited as safety-related. So per Section III, everything still meets, right. And that's why we're saying we're designing it to, so, per Section III, I will do a pressure test to ensure that it holds leak-tightness. That's how I'll be able to certify it or show that I followed the intent of Section III.

MEMBER KIRCHNER: Yes, exactly.

DR. DORON: But we're not crediting it for safety-related seal.

CHAIR PETTI: So I think we ran into this yesterday. At least in my mind, you know, okay, is it safety-related, is it not. And in my mind, not safety-related moves you to a different part of the code; but, in fact, they're still staying inside Section III, they're just not crediting it. So it's a nuance that I hadn't appreciated until we explored it yesterday.

MEMBER REMPE: What about operational

testing? I mean, do you have to check if it's safety-related every so many months for maintenance to ensure it's still leak-tight and, because it's not safety-related during operation, it doesn't have to have that type of --

CHAIR PETTI: But if it's a Section III --

MEMBER REMPE: They still have to do testing during operation --

CHAIR PETTI: They have to follow --

MEMBER REMPE: -- throughout its four-year life --

CHAIR PETTI: Tied to Section 11, Section 11.

MEMBER REMPE: So for its whole four-year lifetime, they have to do this. I apologize. I had to go to other meetings for some of yesterday, so perhaps you discussed this yesterday, but that would be my concern.

DR. DORON: Let me just make one statement, that Section III, once you build the vessel, there are no requirements for further testing. Once you certify the vessel, there are no more requirements for further testing.

MEMBER BALLINGER: Section 11 doesn't?

MEMBER KIRCHNER: Yes, Section 11 applies.

DR. DORON: We haven't committed to Section 11.

MEMBER BALLINGER: Ah.

MEMBER REMPE: That's a big difference.

MEMBER BALLINGER: That's a big difference.

MEMBER KIRCHNER: Yes, yes. Because that would be a concern, Ron. I mean, the vessel itself is credited for safety function, so you would inspect it accordingly, and that would be Section 11.

DR. DORON: Well, we will inspect the vessel for the vessel itself will not leak. It is credited, okay. So what we're talking about is above the fluid level is what we're discussing here. We're talking about the gas space. But then the fluid level, we are crediting it and those welds will be inspected.

MEMBER KIRCHNER: This is an interesting precedent, Ron, from an application to a primary system. We discussed a lot of this yesterday, but okay.

MEMBER BALLINGER: It will be interesting to see how that works out in the operating license.

DR. BLEY: I have to admit -- this is Dennis Bley -- this is a bit confusing to get the implications of this. But why have you decided to go this way?

DR. DORON: For simplicity. I mean, we could talk to -- do you want to answer? You want to talk, Matt? Okay.

MR. DENMAN: This is Matthew Denman, the radiological source term lead. In our transient analyses, we do not credit the vessel head for radionuclide retention. So once radionuclides reach the gas space, likely via evaporation from the FLiBe-free surface, there is no hold-up in the vessel and the radionuclides quickly move to or seem to move into the building and then out to the environment. And because our safety case, our functional containment approach, allows us to not credit the vessel head for radionuclide retention, we are able to do the approach that Oded just mentioned.

MR. GARDNER: So this is Darrell Gardner from licensing. I just also wanted to add I know there's a lot of talk about precedent, but, to my knowledge, we really haven't licensed an advanced reactor that's not an LWR like this yet. So I would caution us to be careful about precedent because I'm not sure that there is. We have to look at this technology as it's being presented, and, for this technology, we do not credit the reactor vessel as a fission product barrier, and that's an important distinction from LWR technology, which does credit reactor vessel and primary piping as a fission product barrier. That's the distinction that matters here and why you see things that might seem unusual compared to LWR technology.

MEMBER KIRCHNER: Well, there have been advanced reactors, like Fermi, that have been licensed.

CHAIR PETTI: Thank you. I was going to say that.

MEMBER BALLINGER: I would caution that we don't get, in this case, because of all these precedents, hidebound by the rules and don't use common sense and Occam's razor for some of these designs. If that's meant to be cryptic, you're right.

CHAIR PETTI: Let's keep going.

DR. DORON: Okay. The diagram on the right highlights the layout of the upper head. I'm not going to step through every single one of those unless you want me to. Do you want to pause on it for a second, or do you want me to walk through those?

CHAIR PETTI: Silence means keep going.

DR. DORON: Okay. Very good. Let's go to

the next one then.

MEMBER REMPE: Just a second. Again, I keep bringing up the coolant level sensor. Have you made any progress on what that sensor will be? I don't think it's documented what you're going to use in the PSAR, right? Do you want to wait until Chapter 7 to discuss this?

DR. DORON: Well, we've made progress, but the details will come later.

MEMBER REMPE: Thank you.

DR. DORON: Yes. So the internals are compromised of the core barrel, which I mentioned previously. It is also 316H. The downcomer is part of the normal natural circulation flow paths. Recall from what I was saying that the downcomer is formed by the concentricity of the core barrel and the vessel, the gap between those. The reflector support structure is 316H. It defines, it helps define the flow path into the core and supports the reflector blocks. You can see it on the right diagram there at the bottom of the vessel indicated as the reflector support structure.

The reflector blocks are ET-10 graphite. They are going to be qualified per the topical, the graphite topical KP-TR-014. They form the fueling shoot, flow channels, core, upper plenum, hot well, bump well, the fueling shoot, and diode pathway. They're part of the normal circulation flow path. The diode pathway is in the reflector block and is also 316H. It's within the blocks, and they're part of the natural circulation flow path, as well.

CHAIR PETTI: So question. How is the core barrel supported.

DR. DORON: It is attached at the upper portion of the vessel.

CHAIR PETTI: Okay. At the top. Okay. Thank you.

DR. DORON: Yes, sir. There's a secondary metallic hold-down structure that precludes damage to the natural circulation flow path from a postulated air ingress event.

Go back one real quick. So the diagrams at the right there are from PSAR directly, and they just highlight all the components there that I was discussing. There's a section cut through the diode pathway. You can see that as section cut A. This is going to be important for when we go into the flow path discussion here I think on the next slide.

MEMBER HALNON: This is Greg. How big are the clearances are we talking between the graphite reflector and the core barrel?

DR. DORON: We did not specify that.

MEMBER HALNON: Okay. I guess the concern would be, if there's any vibration or anything to that, degradation in the graphite through contact.

DR. DORON: Yes, I mean, that's being considered. I'll say it's relatively tight. There is FLiBe that takes up any space that's not physically filled by the graphite.

MEMBER HALNON: Okay. So it's hydraulically buffered in some ways.

DR. DORON: Correct.

MEMBER HALNON: Yes, okay. All right. Go on. Thanks.

DR. DORON: Yes. So we've got two diagrams at the right here, one indicated being the normal flow path and one indicating the natural circulation flow path, so we'll talk through this one here. So the natural circulation flow path, or what we call force flow, essentially what I'm stepping through here is the process of the arrows. So we have the cold leg, the downcomer, the reflector support structure --

MEMBER HALNON: I'm sorry. Are you talking to the one on the left or the right?

DR. DORON: The one on the left first, the natural circulation path first.

MEMBER HALNON: But that says normal operation coolant flow.

DR. DORON: Apologize. It's early here. Forced flow coolants -- sorry. Normal operation, the one on the left. Thanks for catching that.

So the normal circulation path: the cold leg downcomer, reflector support structure into the reflector, the coolant inlet channels, the core, the coolant outlet channels, PEM, hot well, the pump weld, primary salt pump, pot leg, and then the heat is removed by the heat reduction radiator. So what I stepped through there is the process of the flow going all the way through the system.

Next, I'll discuss the figure on the right, the natural circulation flow path. This is during postulated events. I have the cold leg, really the downcomer, the reflector support structure, reflector, so the coolant inlet channel, the core, coolant outlet channel, the PEM, the hot well, the pump well, the natural circulation pathway to the fluidic diode, back to the cold leg, the downcomer, and the heat is removed from the vessel wall by the DHRS. And I want to pause here to make sure that this is absorbed.

MEMBER HALNON: So the fluidic diodes, this flow, is it just the pressure on the normal operation that keeps it from flowing? Because it looks like the flow could go in that direction on the normal operation coolant flow path.

DR. DORON: Correct. Yes, there is bypass, there is some bypass there during normal operation, and that's part of the temperature monitoring that we do on the diode.

MEMBER HALNON: Okay. But the primary way of keeping that flow is the pressure on the coolant.

DR. DORON: Is the pressure differential on the diode.

MEMBER HALNON: Okay.

MEMBER KIRCHNER: Could you further describe the diode? It looks like a check valve kind of function. Can you just share a little about the physical --

DR. DORON: We provided very little detail on the specifics of the diode and the PSAR. It's active work, and the details will obviously will be forthcoming in the OLA. I apologize if that's less than satisfying.

MEMBER BROWN: But, conceptually, it looks like the pressure of the incoming flow is going to be what keeps the flow up through the core from going back toward the natural circulation path. So whatever it does, it's going to have something to do with higher pressure on one side and lower on the other. Is that a rational assumption?

DR. DORON: That's a rational assumption.

MEMBER BIER: Hi. I have some questions. This is Vicki Bier. And these are very high-level questions coming because I'm a PRA person, I'm used to everything works because you have a bunch of bumps and valves, so it's not, you know, a specific challenge to anything in the design but just to educate me.

So first of all, are you kind of relying on the Hermes demonstration to know that the natural circulation is going to work, or you are 100-percent confident before the demonstration that this is going to work?

MEMBER BROWN: That's a long pause.

DR. DORON: Well, we will be doing, I mean, we are doing testing. I can't recall if we've committed to it or not.

MEMBER BIER: Okay.

MEMBER KIRCHNER: I think it's in Appendix A, Oded, isn't it? In the list of, I don't know if you call it R&D, but supporting work for the PSAR.

CHAIR PETTI: I think you're right, Walt.

MR. PEEBLES: This is Drew Peebles, senior licensing manager. So we did get an RAI on the specifics of testing, monitoring, and inspection of the fluidic diode as that part of the natural circulation pathway, and we did commit to those items in that RAI response. I would characterize that a little differently than finding out if it's going to work in the demonstration reactor. We will provide assurance that it is going to be a functional natural circulation path before operation. It will demonstrate the function as we're demonstrating the rest of the technology in the test reactor.

MEMBER BIER: Okay.

CHAIR PETTI: But on initial start-up, can you do a test to test that?

DR. DORON: I can talk to that a little bit. So there's monitoring, there's temperature monitoring, and then there's also -- maybe if we go back to the top head picture. One more. Yes. There's four -- where are they here on this one? The green, there's four green circles there, yes, that are being highlighted right now. Those are nozzles that allow for inspection and potential testing of the diode. We didn't specify details there, but that's what those nozzle ports are for. So there will be a way, there will be multiple ways for us to ensure functionality before we start up.

MEMBER BIER: Okay. Another follow-up. You talked about preventing air ingress from disrupting the natural circulation. Are there other things that might challenge that, whether it's some kind of debris build-up or hot spots that you hadn't anticipated or whatever? Just, you know, what have you thought through in that regard?

DR. DORON: Yes, that's a good question. I don't believe that I said that air ingress would disrupt the flow. If I did, I misspoke. So I'm not concerned about that. Hot spots, that shouldn't affect, even pretty dramatic differences shouldn't affect this process.

As far as debris, that will be precluded through design. And, again, we'll be able to monitor these, and so we will know if their performance has been degraded.

MEMBER BIER: Okay. Thank you.

DR. DORON: Yes.

MEMBER KIRCHNER: Oded, are you going to try and ensure in your design approach that you have continuous circulation through the diode? Because what you don't want is a cold spot or, as Vicki suggested perhaps, a place that graphite or something else might accumulate and have a deleterious effect on that function.

DR. DORON: Yes, I can't remember who mentioned it a few minutes ago, but you correctly identified that there will be, let's say back flow during natural circulation and that back flow will actually be utilized through our advantage through temperature monitoring.

MEMBER KIRCHNER: No. Back flow during normal circulation.

DR. DORON: Sorry. Yes, I keep flipping it. I apologize. Thank you for catching my mistake. During normal operation. And that's what we will use for temperature monitoring, and so that will allow us, again, to see any degradation in performance, which would be indicative of the things you just mentioned there.

MEMBER HALNON: But that would also lock it open in the wrong direction. I mean, so I guess the design will be important. That's --

DR. DORON: The design is important and, again, you will see that through the temperature monitoring. If you start having excessive flow, you will know that.

MEMBER HALNON: Well, I'm more concerned with it being locked open and needing natural circ and it not flowing in the correct direction. Keep in mind we're thinking check valve, you know, with a flapper, so it may be different in that respect. Maybe there's not a mechanical issue. So that's why I'm saying it will be interested in the --

DR. DORON: The detail I can say is it's not a check valve.

MEMBER HALNON: Okay. We'll have to change our paradigm of thinking how that's going to work.

MEMBER KIRCHNER: Well, Oded, this is Walt Kirchner again. You know, our job is not to help you design this, but it would seem to me that one of the things you would consider is just taking the loss of having a constant bypass so that you don't have any concern that it's not going to be functional when the demand comes. Just an observation because with these kind of systems, as you know, you're going to go to extensive effort to prevent freezing and other plugging kind of issues for a salt-like system. So you may just take, intentionally take a loss so that you can guarantee the safety function is available when the demand comes.

DR. DORON: Yes. So, I mean, you're bringing up a great point, and I appreciate the help in the design. So we certainly are, even in this design, we are planning to utilize that bypass, so we are using it to our advantage. So, certainly, I mean, you're 100-percent on the right track with our thinking.

Okay. There are a lot of words coming up on the next two slides. I will apologize. The reason is because the vessel is a critical part of the safety system, and so it does satisfy a lot of PDCs. So please bear with me.

The reactor vessel reflector 316H structures are designed, fabricated, and tested consistent with ASME Section III, Division 5 standard. It satisfies PDC 1.

The reactor vessel, the reactor vessel internals, reactor vessel attachments, are classified as SDC-3 per ASCE 43-19 to protect against failure during design basis earthquake. This satisfies PDC 2.

The reactor vessel and vessel internals design accounts for environmental and dynamic effects, like thermal expansion of the vessel shell and bottom head, mechanical loadings from static weight, and forces from the pebble bed, coolant and core components during start-up, normal operation and postulated events. This satisfies PDC 4.

The reflector block design maintain a geometry in coolant flow path to ensure that SARRDLs will not be exceeded by supporting coolant flow through the reflector via gaps and flow channels, thereby cooling the reflector and maintaining its structural integrity and the integrity of the coolant flow path. This satisfies PDC 10.

The reactor vessel is fabricated and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture, and the vessel material is qualified in accordance with the metallic material qualification topical report. The vessel is operated within as-designed operational and transient conditions and monitored for changes during in-service inspection and testing. This satisfies PDC 14.

The reactor vessel is fabricated and tested to ASME standards. The reactor vessel design supports pre- and in-service inspection and catch basins with sensors are used to detect leakage. This satisfies PDC 30.

CHAIR PETTI: So just a question. When you refer to the in-service inspection, you're talking about only the pass to the credit?

DR. DORON: Correct.

CHAIR PETTI: Thank you.

DR. DORON: Think below the fluid level.

CHAIR PETTI: Right.

DR. DORON: Okay. Onwards and upwards. The reactor vessel design has margins that withstand stresses under operating, maintenance, testing, and postulated events by precluding material fatigue, thermal, mechanical, and hydraulic stresses that would degrade the reactor coolant boundary. Stress rupture factors encompass transient conditions and leak-tight design of the reactor vessel head to minimize air ingress. The design prevents fracture of the reactor coolant boundary. This satisfies PDC 31.

Reactor vessel design permits inspection and monitoring of the structural integrity and leak-tightness of the reactor coolant boundary using the material surveillance system, MSS, to confirm irradiation assistance, stress corrosion/cracking that is non-existent or manageable. This satisfies PDC 32.

The core valve design maintains reactor coolant inventory in the events of a break in the primary heat transport system using anti-siphon cutouts on both sides of the core barrel. This satisfies PDC 33.

The flow path established by the design of the reactor vessel internals support the removal of residual heat from the core to ensure SARRDLs are not exceeded during normal operation and postulated events. The physical geometry and structure of the reactor vessel internals provide a pathway for force flow and continuous natural circulation. This satisfies PDC 34.

The fluidic diode, reflector blocks, and downcomer are designed to maintain their structural integrity in order to establish a flow path for continuous natural circulation during a postulated event. The passive cooling of the reactor core prevents damage to the vessel internals due to overheating and, therefore, ensures the total geometry of the core is maintained. This satisfies PDC 35.

The functional capability of the natural circulation flow path is confirmed during normal operation by temperature monitoring. Appropriate periodic inspections of fluidic diode are performed via head penetrations. This satisfies PDC 36 and 37.

Finally, the reflector is qualified to

maintain its structural integrity to support residual heat removal in accordance with the graphite material qualification topical report. The reactor vessel is classified as SDC-3 and will maintain its geometry to support the insertions of shut-down elements. And this satisfies PDC 74.

I believe that's all I got.

MEMBER KIRCHNER: Oded, could you give us a little bit more detail on the cutouts, that was hard to see in the PSAR, that serve the function of the breaking, you know, a flow path that would drain the vessel.

DR. DORON: I don't think we added that detail because that's a pretty specific design detail, but what I'll tell you is they're in the upper region of the core barrel and, therefore, as you lose fluid, you would suck in air and break the siphon.

MEMBER KIRCHNER: Okay. All right. Thank you.

DR. DORON: Yes.

CHAIR PETTI: That's your last slide, right?

DR. DORON: Yes, sir.

CHAIR PETTI: Yes. So let's hear from the staff.

DR. DORON: I appreciate your time on that.

MR. CHERESKIN: Good morning. This is Alex Chereskin from NRC staff, Division of Advanced Reactors. I'll be -- sorry. This good? All right. Thanks.

So like I was saying, this is Alex Chereskin from the NRC staff, the Division of Advanced Reactors, and I'll be presenting the staff's review of Section 4.3 for the reactor vessel system.

Next slide, please. So the staff's presentation is just going to touch on a very brief overview of the PSAR. I'll probably try and keep that detail limited because we just heard from Kairos for a lot of that. We'll touch on the regulatory basis and the PDCs included in our review, the reference topical reports in Section 4.3 of the PSAR, as well as the staff's technical evaluation, our conclusions, and then our regulatory findings.

Next slide, please. Okay. So as we just heard from Kairos, the reactor vessel system has a vessel shell, top and bottom heads, as well as all the internals which are listed here on the screen. And so it includes things like the reflector blocks, the fluidic diodes, the barrel, and the reflector support structure. The purpose of the reactor vessel system is to contain the core and provide for the circulation flow path for the reactor coolant pebbles and also to allow for the insertion of reactivity elements. And the last thing I note here is that, you know, the reactor vessel system is a safety-related system.

Next slide, please. Okay. So this slide contains the regulatory basis for the staff's review. I am not going to go into too much detail here. As you can see, there are a lot of principal design criteria listed on the slide, and we have the criteria from 10 CFR 50 and these are generally related to the information that needs to be provided for issuance of a construction permit. And I'll touch on the PDC a bit more in the subsequent slides, but, in general, they're aimed towards ensuring that appropriate codes and standards are used commensurate with the safety significance of these components, ensuring that the vessel system is structurally sound and can withstand the environments in which these components are going to operate, you know, to ensure the vessel system can maintain its integrity to support the passive residual heat removal and insertion of reactivity elements. And there are PDC that also relate to allowing for inspection and testing, as appropriate.

Next slide, please. So in Section 4.3 of the PSAR, Kairos referenced a couple of previously-approved topical reports from the staff. So the first topical report that was referenced is the topical report which describes the principal design criteria for the Kairos power design, and the second topical report is for the metallic material qualification, which covers 316H materials in environments representative of what would be expected in the design and also the graphite material qualification report for the ET-10 graphite that Kairos plans to use. And on this slide here, I don't have dash A for those topical reports, although they were recently issued, I think, within the past couple of weeks, so those are approved topical reports.

Next slide, please. Okay. So this slide covers the staff evaluation for Principal Design Criteria 2, which would require safety-related system structures and components to withstand effects of natural phenomenon. And so in the PSAR, Kairos described the seismic interactions that would be considered to ensure the vessel system can withstand a design basis earthquake. The staff has reasonable assurance this will be met because the vessel, the internals, and the attachments are classified as seismic design Category 3, as per ASCE 43-19, and also because the design will ensure that safety-related SSCs would be protected from failure of nearby non-safety-related SSCs. And this is also consistent with the criteria in NUREG-1537, which would require the vessel to be able to withstand all mechanical and hydraulic forces and stresses to which it could be subjected to in its lifetime.

Next slide, please. So this slide is the staff's evaluation of PDC 4, which requires protection against environmental and dynamic effects. And so the PSAR states that the vessel system can withstand internal and external static and dynamic loads during normal operations and postulated events. And, you know, the staff has reasonable assurance of this because the vessel system is designed to account for these loads, which would include static weight, seismic loads, and forces from the pebble bed, coolant and core components, you know, pipe whip hazards. We review that to ensure that those are not a concern during the operating license stage. And this also helps meet that NUREG criteria that I mentioned on the last slide.

I would also mention that Kairos has stated the vessel system would be in accordance with

Section III, Division 5, of the ASME code.

Next slide, please. Okay. So this slide covers the staff evaluation of Principal Design Criteria 10. And so PDC 10 requires core heat removal to have appropriate margin, and the role that the reactor vessel system plays in satisfying this PDC is mainly to ensure the flow path needed for the adequate core cooling. This would be accomplished by ensuring the integrity of the reactor vessel system, so you have the vessel and the internals that form the flow paths that Kairos described during their presentation. And there's reasonable assurance that this will be achieved because, as noted here, there are the two material qualification topical reports that will cover the conditions those materials are expected to encounter in this design.

And, additionally, Kairos will demonstrate the compliance with Section III, Division 5, you know, the appropriate sections, Section III, Division 5, for either stainless steel or graphite components. And so, you know, that, combined with the vessel and internals and attachments and being classified as SDC-3, in order to account for those dynamic behaviors to make sure that they can operate during a design basis earthquake, you know, and also combined with protecting these safety-related components from the failure of nearby non-safety-related SCCs.

I would also note that, for maintaining the coolant flow path, that Division 5 design rules for graphite would require the consideration of dimensional changes from things like thermal expansion and radiation damage, which, you know, that would allow the graphite components to ensure that the flow path is maintained even, you know, through the spectrum of thermal and irradiation conditions that they're expected to experience, and this is consistent with the NUREG-1537 criterion that graphite components would have to accommodate radiation damage and thermal expansion.

In addition to that, you know, the graphite qualification topical report also discusses the effects of graphite oxidation, and so that would also be covered to ensure that graphite integrity is maintained or satisfied as PDC.

Next slide, please. So this slide covers the staff evaluation of PDC 14, which requires safety-related elements of the coolant boundary to be designed and fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. And so the staff has reasonable assurance this will be met because, as stated here, the vessel system is going to be designed for operational and transient stresses with an appropriate load methodology.

And, additionally, as mentioned before, the two materials qualification topical reports will have testing to quantify effects of applicable degradation mechanisms for 316H components, as well as the weld filler metal, and as well as the graphite components, as well. And so, you know, one thing that the staff will look for at the operating license stage is the results of these qualification programs to ensure that these degradation mechanisms aren't too significant to prevent meeting PDC 14.

In addition, the vessel system is designed to permit appropriate periodic inspections and contains features such as catch basins to detect any leakage. And these will also, for the confirmation, the inspection will help for confirmation of degradation rates and, you know, ensure potential corrective actions are taken to ensure that the safety-related parts of the coolant boundary are maintained.

In addition, at the OL stage, the staff would review those inspection and monitoring programs, and this is all consistent with the NUREG-1537 criterion that the vessel and the coolant have chemical compatibility. That would be demonstrated through the metallic material qualification topical report. And also the Division 5 rules would help to meet this PDC, as well, to ensure the integrity of the safety-related components of the boundary.

Next slide, please. Okay. So this slide groups together the evaluation of PDCs 30 and 31. PDC 30, which would require the components that are part of the coolant boundary to be designed, fabricated, erected, and tested with quality standards commensurate with their safety functions, and so that's satisfied or is consistent with using ASME Code Section III, Division 5, for design fabrication, those aspects of the code. And then Principal Design Criteria Number 31 requires those safety-related elements of the coolant boundary to have sufficient margin to ensure that when they're stressed under all conditions so that the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. And so the staff has reasonable assurance that these will be met because, as noted, Kairos is going to be using Section III, Division 5, and that covers effects like high-temperature creep and fatigue for the metallic components in this vessel system. And we noted previously, there's also the catch basin that can detect leakage from the safety-related parts of the coolant boundary. The metallic material qualification topical report contains testing to extend the weld filler metal qualification, as noted in the PSAR. And there is also a proposed safety limit in Chapter 14 for the vessel temperature, and, at the operating license stage, the NRC staff would look to ensure that the times and temperatures for metallic components in the system, safety-related metallic components in the system are consistent with the NRC staff-endorsed ASME Code Section III, Division 5.

And, additionally, the topical reports that I mentioned also look at the effects of things like radiation and coolant composition, you know, for both the metals and the graphite materials. And so this would also allow the staff at the operating license stage to ensure that appropriate coolant purity limits are included with the operating license based on material testing that Kairos is going to perform. And so also at the operating license stage, we would look at the results of this testing to ensure that they have the appropriate design margin that's discussed mostly in PDC 31.

Next slide, please. Okay. So this slide is the staff evaluation of PDC 32, which will require those safety-related components that are part of the coolant boundary to be designed to permit periodic inspection, monitoring, or functional testing, as well as an appropriate material surveillance program. As noted here, the PSAR states that the Hermes design will have coupons, component monitoring and inspection, as appropriate. And as I noted before, the staff is going to review the details of inspection and monitoring programs at the OL stage, and so the staff has the reasonable assurance that the vessel system will be consistent with PDC 32 because they do have these coupons, as well as the ability to perform in-service inspection and in-service testing. This is also consistent with the guidance in NUREG-1537 to assess the irradiation of vessel materials because these coupons, as noted in the Kairos presentation, would be used to look at the effects of irradiation on corrosion. And so the staff is going to look at the results of the qualification testing in conjunction with the surveillance and monitoring at the OL stage, as well as the modern inspection programs.

Next slide, please. Okay. So this slide

has the staff evaluation of PDC 33, which is required to protect against small breaks and maintain FLiBe inventory. And so most of the evaluation for this PDC is contained in Chapter 5 of the SE where it discusses aspects, like the anti-siphon features. And so if we have any questions on this, I might just ask that we wait until Chapter 5 of the SE because that's really where it's contained.

Next slide, please. So here's our evaluation of PDC 34, which requires a system for residual heat removal. So as stated in the PSAR, the vessel internal design supports the decay heat removal, and staff has reasonable assurance that's going to be consistent with PDC 34. As you can see, the design maintains the pathway for the coolant, and, you know, the components in the natural circulation pathway are qualified in those two previously-discussed topical reports. And also, as previously discussed, you know, they'd be designed and fabricated to ASME Code Section III, Division 5 requirements. And this combines to give the staff reasonable assurance that these components can maintain integrity and form the pathway needed for coolant flow in both normal operation and natural circulation and postulated events.

And then the other aspects of decay heat removal are evaluated in various sections of the staff's safety evaluation, as noted here. We have thermal hydraulics, the decay heat removal system, and anti-siphon features in different sections.

And the next couple of slides are going to talk about inspection and testing for these components. So next slide, please.

Okay. So these three PDC are pretty similar, so we just kind of condensed them down to one slide here. With PDC 35 requirement passive residual heat removal system to ensure cooling during postulated accidents, you know, and PDC 36 requiring the appropriate periodic inspection of important components and PDC 37 requiring appropriate periodic functional testing. And so the staff has assurance that these PDC will be met because, again, as noted, we have the topical reports that have the qualification methodologies to bound the expected conditions that these components are going to see in the Kairos design environment, as well as the temperature monitoring that was described in Kairos's presentation, and as well as the inspection, the ability to inspect the fluidic diode device that I think Kairos also went into a little bit of detail in their presentation, as well. And this is one of those things that it's safety-related, it's an important component, and so the ability to be able to inspect the diode, I think maybe that also goes to the conversation we were having earlier about potential graphite particles, for example, and so the ability to be able to inspect that gave the staff some reasonable assurance that, you know, if you were seeing maybe a build-up of that particles, that you'd be able to take corrective action if needed.

And I would also note that, you know, at this stage, at the CP stage, we do not have the details of the inspection and testing programs, but those will be provided with the operating license. And so the staff will be able to review them at that time in conjunction with the overall design to ensure that, you know, their component integrity plan is able to ensure that they can inspect and test what they need to in the reactor vessel system.

CHAIR PETTI: So I don't recall seeing details. The inspection is done after it's shut down. It's not in situ while the reactor is operating. Of the diode because it's, you know, it's inside.

MR. CHERESKIN: We did not have the details of exactly, I'll say timing of the inspections. I'm not sure whether it would be possible to do while FLiBe is in the system or not.

CHAIR PETTI: I mean, you know, good FLiBe is clear, but as soon as you get impurities it's cloudy, so that would potentially rule out anything in in situ.

MR. CHERESKIN: Yes. Next slide, please. Okay. And so this slide details the staff review of PDC 74, which is discussing the requirements to be able to maintain that natural circulation flow path, as well as to allow for the insertion of reactivity elements. And the staff has reasonable assurance the vessel system will be able to meet that. Then I'll bring up those two qualification topical reports to ensure the materials can withstand the environment that they're going to be subjected to.

In addition, Section 4.3 of the PSAR has graphite thermocouples, which would also allow the staff the assurance that the graphite is going to be within its temperature conditions because the temperature at which graphite is irradiated obviously has a great effect on its properties and its ability to maintain its geometry, which would be important to allow for the reactivity element insertion.

And, in addition, the Division 5 rules for both metallics and graphite help to provide the staff assurance that this is going to be met because it's going to, especially for the graphite, require consideration of a lot of these dimensional changes as the properties change over time and with different temperatures to ensure that the appropriate allowance is there for the reactivity elements to be able to be inserted.

And so at the operating license stage, you know, we look to review the final design of the graphite components specifically. And this is all also consistent with the NUREG-1537 requirements that require graphite components to be compatible with their chemical, thermal, mechanical, and radiation environments.

CHAIR PETTI: Just a question or comment. The testing will, you know, give you shrinkage rates with temperature and fluence. Again, remember, these are little small-sized tests. There's also this jump from that to the full component where there's gradients in temperature, gradients in fluence, so there's a calculational aspect here, right, that you get the assurance not just from the testing but it's through all that analysis in Div 5, right?

MR. CHERESKIN: Right. Division 5 is

obviously very important in that end. The design portion of Division 5 requires that you would account for those gradients. It specifically calls out thermal stresses generated by those temperature gradients, as well as the radiation damage that can cause internal stresses to the graphite, and that all feeds into the probability of failure calculation, which looks at, you know, the initiation of cracking essentially. And we'd obviously look at those results in combination with, like, the concept of damage tolerance and graphite to determine whether or not the cracks might occur in an area where it could impact one of those functions.

Next slide, please. So this slide, I think, is a little bit of a repeat. As the PSAR doesn't contain the details of the testing and inspection program to the reactor vessel system, and so that's going to be something that the staff will review with the operating license application.

DR. SCHULTZ: Alex, I'm glad you got this page on testing and inspection, but you've mentioned clearly that this is a very intense evaluation by the staff at the operating license. Are you confident that you and Kairos are on the same page with regard to the programs that they're going to be submitting in advance of the operating license? You've had a lot of interactions with them, not directly on these areas but certainly discussing that things need to be done in a certain fashion and so forth. Do you feel you're comfortable with what will be coming in in advance of the operating license?

MR. CHERESKIN: I would say I think so. I think that's part of our review for the construction permit is to have that assurance that we think it can be done, and so, from that perspective, yes. I would also note that you're right. I think there's a lot of work that still needs to be done. We'd still need to look at the specifics of how some of this stuff can be done, especially if you're in a new environment with novel components.

And so I think that's an appropriate focus area, but, you know, there have been a couple of times where even Kairos has noted that they have ongoing research and development, and I think we need to see the outcome of that.

DR. SCHULTZ: Very good. Thank you.

MEMBER REMPE: So I've heard sometimes that changes were made in the design because of your interactions with the folks from Kairos. Were any changes made because of your interactions with them on this topic?

MR. CHERESKIN: I think one example, you know, we issued the RAI on the inspection for the fluidic diodes, and I think that was part of the change to have those inspection ports, to be able to have the ability to inspect those devices, if needed, as appropriate.

MEMBER BALLINGER: This is probably more of a question that eventually, for the FHRs, will be important. But with respect to Section 11, are they using Division 2? Are they going to have a RIM or whatever, I forget the acronym.

MR. CHERESKIN: I don't believe they've committed to specifics of that. Meg, do you happen to recall?

MS. AUDRAIN: They have not committed to using a RIM program yet. Meg Audrain, NRC staff.

MEMBER BALLINGER: I mean, this is a small system and everything, but to get engaged between those two divisions might have implications for the FHR, the bigger points.

CHAIR PETTI: So just another comment, and I think we'll get back to this in Chapter 5. But as we're hearing about the downstream implications of functional containment and the ability to classify what's credited and what's not, given this is first-of-a-kind, what I haven't heard is, well, you know, okay, we understand that logic, but this is first-of-a-kind from defense-in-depth. Wouldn't that be a good idea? Common sense. So I think we'll be back on this when they get to Chapter 5. If we were in Part 53 and Framework A, there's that defense-in-depth look that you look at everything. And even though the logic drives you over here, you come back with defense-in-depth and you go, well, you know, no, we should beef up this and beef up that. That's not here because of the strategy, but that's not a bad, that's an important part of Framework A, I think.

MR. CHERESKIN: Thanks. Next slide. Oh, sorry.

MEMBER BIER: A question or a comment to Dave's comment, which we discussed yesterday, as well, that it's a challenge doing these separate construction permanent operating license because there's certain detail that you might want to see that a licensee doesn't have to provide at this point. And I'm just curious, do you have a sense at the level of detail available was sufficient for you to perform your review with reasonable assurance, or are there things that really are kind of nagging doubts because of not having certain things spelled out? Sorry. My mike was off. I don't know if I should repeat that all. Okay. Great. Thank you.

MR. CHERESKIN: So the staff does have the reasonable assurance for issuance of a construction permit, and I think, as we've discussed plenty of times, there's stuff that we would like to see at the operating license application. But for a construction permit, you know, our safety evaluation documents, that we have what we need.

MEMBER BALLINGER: This is not a reflection on the staff, but, again, this is a first-of-a-kind system, and is there any thought, we have this set of rules and we have a tendency to follow the rules, that's it, and we think that's going to be fine, but is there any thought to, for lack of a better word, having what we've termed in the past a murder board? That is to say, a separate group that takes a look at what you've done for the overall package that said, you know, let's look at this and see if there's some areas where, in spite of the rules or because of the rules or whatever, that we should probably take a closer look because this is a new thing and we're establishing precedent here? Is there any thought on a high level been given to that kind of, and I guess I'm pointing --

MR. SCHMIDT: This is Jeff Schmidt from the staff. I would have to say, no, we haven't really thought of that aspect. I mean, it may be more appropriate for, like, the commercial design --

MEMBER BALLINGER: Well, that's what I mean --

MR. SCHMIDT: -- than the test reactor.

MEMBER BALLINGER: -- you're establishing a kind of a precedent here, and the commercial design is going to be much more -- yes.

MR. SCHMIDT: Yes. So, no, we haven't, I don't think we've really considered that aspect. I mean, we've tried to do it internally just within the staff but not another external body beyond --

MEMBER BALLINGER: I don't mean external.

MR. SCHMIDT: -- the normal process.

MEMBER BALLINGER: I mean also within the --

MR. SCHMIDT: Oh, you mean like a separate staff --

MEMBER BALLINGER: Yes.

MR. SCHMIDT: -- like an internal review. No, I don't think we've really considered that.

MR. BEASLEY: So this is Ben Beasley of

the staff. And Jeff's correct, we didn't consider that. It's not a bad idea, but I will admit that Jeff's examination and Alex's examination and Michelle's and the entire core team's look was robust, was thorough. You know, Jeff is very knowledgeable and did, you know, think through what are all the things that could go wrong. And so I do have confidence --

MEMBER BALLINGER: Remember, this is not a criticism.

MR. BEASLEY: No, no, I understand.

MEMBER BALLINGER: I don't doubt for a millisecond that it's robust and all the things that you're saying, but this is a new system.

MR. BEASLEY: Yes, so it is a good idea.

MEMBER BALLINGER: And so, you know, sometimes you read the same thing over and over and over again and you miss the misspelling over and over again.

MR. BEASLEY: Well, and so I guess the aspect I was thinking is that we were not taking Kairos's word for this is all that's needed. You know, we were considering, you know, what is all that's needed, you know, for our independent review.

MEMBER BALLINGER: There is a murder

board. It's called the ACRS.

MR. BEASLEY: And you guys do a fine job at that.

CHAIR PETTI: Okay. With that, we can move on. Alex.

MR. CHERESKIN: And so just to wrap up here, you know, the staff has the reasonable assurance that the preliminary design information, the PSAR, is consistent with the NUREG-1537 criteria, as well as the applicable PDC that we just went through, and that this information, the PSAR, is sufficient for issuance of a construction permit in accordance with 10 CFR 50.35. And, you know, as noted, a handful of times here, there's further information that we will look at during the operating license application.

And so that concludes my presentation. Are there any further questions?

CHAIR PETTI: Well, with that, let's move on then to the bioshield Section 4.4. Kairos.

MR. SATVAT: Good morning. Can you hear me? It's muted.

MEMBER HALNON: No, we can hear you fine.

MR. SATVAT: Oh, okay. Sorry. Good morning. This is Nader Satvat, senior manager of nuclear design. I'll be talking about the biological shield as Section 4.4 of the Preliminary Safety Analysis Report.

The function of biological shield is protection of public and the work here from release of radionuclide and the impact of radiation. This meets the radiation exposure goals in PSAR Chapter 11. The design of the shield, the primary shield located just outside of the reactor vessel, there is an image that I will show on the next slide, a secondary shield located outside the primary shield and contains the inventory management and primary heat transfer system. Both shields are reinforced concrete. Details on biological shield will be provided as part of the operating license application.

This is pointing out what was described in a previous slide. The reactor vessel is contained within the primary biological shield, which is contained in the secondary biological shield which holds the reactor cavity, the heat rejection radiator, and inventory management system.

CHAIR PETTI: Can you just give us sort of rough dimensions on that?

MR. SATVAT: The dimensions were not provided in PSAR, but those details are being worked out and will be provided as part of operating license application.

CHAIR PETTI: Is it bigger than a breadbox?

MEMBER BALLINGER: It just looks like the same size as the seismic that we qualified building.

CHAIR PETTI: No, no, I mean this is a part of the building. Right.

MEMBER BALLINGER: Well, okay.

CHAIR PETTI: Yes, all of these are in the seismic -- the moat is outside of that second gray box; is that correct?

MR. SATVAT: Right. Did you hear me? I said that's correct.

CHAIR PETTI: Yes, yes, I heard. Thank you.

MR. SATVAT: This is the last slide.

CHAIR PETTI: And that's just concrete or -- oops. It's just regular nuclear concrete, so it's high-density concrete?

MR. SATVAT: The details of the material is being worked on. There is a workstream on understanding the type of material, the degraded shielding approach, and the localized shielding which is missing on this picture. But all of those will be provided as part of operating license agreement, application.

CHAIR PETTI: And it's an area in which personal entry is not anticipated?

MR. SATVAT: During operation, there is no expectation of personnel to be within this shielded area.

CHAIR PETTI: But you may go in when the reactor is shut down.

MR. SATVAT: Yes, that's possible.

CHAIR PETTI: Okay. Thank you. Okay. Staff. Michelle.

MS. HART: Yes, good morning. My name is Michelle Hart. I am from the Division of Advanced Reactors in the NRC staff, and I'm here to talk to you about our review of the biological shield. Next slide, please.

So as was just described, it's radiation shielding for worker protection during operation, as well as protection of the public from radiation shine. It's reinforced concrete structures, as Kairos just described, and it's a safety-related component for structural support and external event protection but not safety-related for the radiological protection.

I did not perform the review of the structural design basis. That would be part of the review in SE Section 3.6.

Next slide, please. So for our evaluation, the staff assessed whether the PSAR provides an acceptable basis for the development of the biological shield and determined if the objectives of the biological shield design basis are sufficient to protect the public health and safety and the facility staff and to assess whether there's reasonable assurance that Kairos will comply with the regulations of Part 20 during Hermes facility operation. We did not determine that they need that at this point because they are in the construction permit. They are not requesting to have special nuclear material on site.

Next slide, please. So we did ask in audit if they had some preliminary shielding analysis because, as you saw, there's not a lot of information in the preliminary design information. They do confirm and we did confirm that they had some preliminary analyses. They did not provide us details on that, but it is clear that they used isotopic data based on fuel and FLiBe sources to help them determine, to help them determine what's going to be in the biological shield materials and the thicknesses and the dimensions.

The evaluation of its performance to meet Part 20, including the shielding analysis, will be reviewed as part of the operating license application. The overall review of the Hermes facility to meet Part 20 will be part of the Chapter 11 analysis in the OL application, as well.

And that concludes my presentation. Are there any questions? I mean, I know you have concerns.

MEMBER HALNON: It's not concerns. It's lots of questions. What just went through my head was they don't know what the materials are, but they can confirm that there is analysis to support PSAR, which means there's some analysis that says this is going to work which means that there's got to be some idea of what kind of materials there are going to be. I almost feel like, when we get to the operating phase or in a Part 52 type review, we're going to be looking at everything all over again because we don't have any detail here.

So I don't know. I guess I just vented a little frustration. It's nothing against you, Michelle. You guys confirmed that the minimum requirements have been met, so that's good.

CHAIR PETTI: Yes, I would just note that, on these smaller systems, there's some leakage, neutron leakage, more than in large power reactors, so gamma and neutron will be potentially more important than other systems. Just something to be aware of.

Okay. If no other questions, then --

MEMBER REMPE: Again, like what Steve was saying, do they understand that there's a lot more risk when you have less detail? I mean, did that come up in your discussions? I mean, you can build the thing and then, I mean, that's the trade-off, and they understand this?

MS. HART: Well, certainly, the staff, we discussed it among ourselves when we were trying to determine the appropriate amount of information that would be required for a construction permit, and I think we did have a few conversations with Kairos about that point, but we didn't push that point. I think they're aware. They may be able to speak for themselves if they would like.

MEMBER REMPE: Again, it's not just in the U.S., again, because we did this thing and the folks from Finland talked about the issues and how they actually had additional reviews before they had the operational license because they kept trying to reduce the risk because that's their way of doing it, and we aren't. And so maybe, as we go forward, we'll learn from the experiences that are happening.

MEMBER KIRCHNER: Joy, just to jump in, this is Walt. The caution -- doing the shielding, sharing a little bit of Greg's feelings, I mean, they can do a good estimate of how much shielding they need now. I think the staff indicated they looked at that. As Dave said, for a reactor like this, you know what the megawatts are. It's a thermal spectrum. You can pretty much ballpark the amount of shielding that's needed to get the dose down to acceptable levels.

The risk here in the shielding, which is a more -- how should I say it? Not an exotic item in the design. Is that should they require more shielding, that has a ripple effect on the structural part of the design, particularly the seismic loadings and the civil structure. So I think Kairos will do this. They're going to have to lean on the conservative side towards what materials and weights and such will be required to achieve the shielding desired and then do the civil structural design accordingly so that they can carry that load, especially with the seismic isolation. So that's the, that's, I think, the more challenging issue here, not achieving a sufficient amount of shielding.

MEMBER BROWN: Hold on a minute. I've got to mute something. Here I go. I'm in a meeting.

MEMBER REMPE: Charlie, I think you didn't realize -- are you trying to ask a question, or you just left your mike on?

Well, I guess I'm reflecting more, I hear this same comment a lot about different aspects of the design and, of course, safety and meeting the requirements is all we all need to do at this time. But sometimes the regulator gets blamed for things later on downstream when something -- there's a lot less information, and I just hope everybody understands.

DR. SCHULTZ: Well, let's be clear, this puts the onus on Kairos to come up with a design that is going to be easily reviewed prior to the operating license review.

MEMBER REMPE: I agree, but then if they do what they think is adequate and the regulator says, well, no, then it causes problems later. But, again, that's not our problem today.

DR. SCHULTZ: It could, but it shouldn't.

MEMBER REMPE: It shouldn't. We hope.

CHAIR PETTI: Okay. With that, let's move on to Section 4.6. We're a little ahead of schedule, so maybe we can get both Kairos and the staff done before our break. Kairos.

MR. SATVAT: This is Nader Satvat, senior manager of nuclear design, which part of that is the core design. I'll be talking about thermal hydraulic design of the internals, which is Section 4.6 of the PSAR.

These two schematics were provided in Section 4.6. I will not go into great detail there, but what I will talk about is the thermal hydraulic design that ensures that the design meets the PDC as laid out for 4.6.

The thermal hydraulic design is a combination of design features which includes pebbles, a source of heat, reactor coolant, reactor vessel and internals, and the primary heat transfer system which ensures the adequate flow of the coolant through the vessel and vessel internals. The thermal hydraulic design uses multiple heat transfer mechanisms between the reactor materials. The details of those are presented in the topical report for both core design analysis and also the transient analysis.

Thermal hydraulic design includes coolant flow path for normal operation and natural circulation. Natural circulation flow path uses a fluid diode which was discussed in Section 4.3 that minimizes the reverse flow, the bypass.

Qualification or functional testing plans for fluidic diode and the test results to evaluate performance will be available as part of operating license application.

STAR-CCM and KP-SAM computer codes are used in thermal hydraulic analysis of the design, which I'll talk about those in the next slide.

STAR-CCM, as described in Section 4.5, is used to perform thermal hydraulic analysis in the core design methodology. It is a steady-state for heat transfer and fluid flow in the form of a 3D porous media model. It calculates the core material temperature, which is used as input to neutronics model. The detail of that methodology is described in KP-TR-017, Revision 1.

CHAIR PETTI: Just a question.

MR. SATVAT: Sure.

CHAIR PETTI: You know, the whole issue of validation of the CFD is new, and I know that the relevant loop inside ASME, thermal hydraulics, has been working on, I guess, criteria or just some guidance on how one might validate CFD. Are you guys aware that you hooked in -- are you ahead of them?

What's going on there? Have you thought about it at all?

MR. SATVAT: Sure. The part of the STAR-CCM that is used for this purpose is 3D porous media approximation. We are developing internal and have developed internal testing, including different levels of modeling to ensure the numbers that we're producing are adequately conservative for figures of merit that are important for safety.

CHAIR PETTI: Okay. I don't know, I haven't kept track with what ASME is doing, but I know that they had a working group on it.

MR. SATVAT: That's correct.

CHAIR PETTI: Yes. Thank you.

MEMBER KIRCHNER: Dave, I would also point out, and I think Kairos might be aware of this, that porous media kind of techniques have been used for dry cask storage thermal analysis, so you might look at what's been done there, as well.

MR. SATVAT: Thank you.

CHAIR PETTI: Keep going.

MR. SATVAT: KP-SAM is a system that performs transient analysis on postulated event, the progression of different scenarios, postulated scenarios for the reactor. It is simplified models to represent the major physical components and describes major physical processes, including fluid flow and heat transfer. It is used to analyze the progression of events. As an example, insertion of excess reactivity, loss of force circulation, and other relevant accidents. The details of KP-SAM is presented in KP-TR-018, Revision 2.

These are the thermal hydraulic analysis of the vessel and internal is going to address the following PDCs. The slide is wordy, but I'll try to be brief. The design provides adequate transfer of heat from the fuel to the coolant to ensure SARRDLs will not be exceeded during normal operation and postulated events. That's PDC 10.

The thermal hydraulic design of the reactor system ensures that power oscillations that could result in conditions exceeding SARRDLs are not possible or can reliably and readily detect it and suppress. This was discussed again as part of 4.5 that power oscillations in FHRs with the size of Hermes is not possible.

Residual heat is removed during normal operation and postulated events, such that SARRDLs and the design conditions of the safety-related elements of the reactor coolant boundary are not exceeded. The reactor transfers heat from the reactor during postulated events, such that the fuel and reactor internal structural damage that could interfere with continued effective core cooling is prevented.

That's my last slide.

CHAIR PETTI: Thank you. Questions, members?

MEMBER KIRCHNER: Yes, Dave. This is pesky Walt again. This is a relatively small reactor, a small power level. Have you done what I'll call a static passive calculation like has been done for the HTGR designs to just assume for whatever reason the fluidic diode does not function and you have a stagnant system? Have you looked at a worst-case scenario such as that and looked at what the pebbles and the core would see in a decay heat removal passive cool-down situation as a bounding event?

MR. SATVAT: We have not done that calculation. Our testing and design process, including start-up, is assessing and continued monitoring of natural circulation through the diode is going to ensure that they perform their function when it's needed for them to perform. So we have not looked at that analysis. However, internally, we do have capabilities to look at those scenarios, but we have not considered them.

CHAIR PETTI: Okay. Then let's turn to the staff. Ben.

MR. ADAMS: Good morning, everyone. This is Ben Adams from the NRC staff.

CHAIR PETTI: Please speak closer or louder.

MR. ADAMS: Good morning. This is Ben Adams from the NRC staff. I will be presenting on the NRC's review of PSAR Section 4.6, thermal hydraulic design. We're going to go over the regulatory basis and PCs quickly, look at the staff technical evaluation and conclusions, and the regulatory findings.

The regulations for the regulatory basis are the same as they've been in the other sections. It's 50.34(a), 10 CFR 50.34(a), 10 CFR 50.35, and 10 CFR 50.40. We reviewed the principal design criteria relevant to the section which are listed on this slide. That would be PDC 10, reactor design; PDC 12, suppression of reactor power oscillations; PDC 34, residual heat removal; and PDC 35, passive residual heat removal. Some of the findings in this section are related to ensuring the design is consistent with these PDC and some of the findings are related to the analytical methods which we thought were important to capture in this section of the safety evaluation. We did also use NUREG-1537 as guidance for this section.

So for the reference topical reports, we used KP-TR-003, which is the principal design criteria topical report, and we also referenced KP-TR-017, which is the core design and analysis methodology technical report. And we looked at KP-TR-018, which is the postulated event analysis methodology.

So I'll give a brief overview of the analytical methods. I know Kairos just discussed this a little bit. The core design methodology included SERPENT 2 and STAR-CCM+, which was discussed a little bit in Chapter 4.5, the presentation yesterday. STAR-CCM+ is a computational fluid dynamics code or a safety code.

The transient analysis methodology includes the use of KP-SAM and KP-BISON. KP-SAM is a modification of the SAM code, which is a transient code; and KP-BISON is a modification of BISON, which is a fuel performance code. We will discuss those in more detail later with the Chapter 13 presentation on April 18th, but we'd like to make clear that these codes have not been reviewed for verification and validation yet, and the NRC has not approved the use of these computer codes.

Moving on further to the analytical methods. The Hermes model accounts for the following heat transfer modes. It accounts for pebble-to-pebble convective heat transfer, pebble radiated heat transfer, pebble-to-pebble heat transfer via a couple contact conduction, pebble-to-pebble heat transfer via conduction through the coolant, and conductive, convective, and radiated heat transfer to the reflector. So, specifically, the staff reviewed heat transfer modes to make sure that they were appropriate, and we reviewed the heat transfer equations given in the technical reports and the PSAR to do this staff-performed literature review, which included checking code manuals, like the SAM and MELCOR code manuals, SAM being what KP-SAM was based off of and MELCOR what was we used for our NRC staff's scoping calculations. We also conducted research. We also discussed this at length during the audit.

Staff reviewed the thermal conductivity equations and the pebble bed pressure drop equations, correlations used, and their applicability. We did not review Kairos's decay heat methodology. I believe it is named in the updated PSAR revision, but we will be reviewing that later at the OL stage.

I would also like to make clear that the NRC is not approving the use of these equations or correlations, and the NRC is not approving the research references in any way used to make these findings. And, again, the NRC has not approved the use of the computer codes. We're saying that it's okay that Kairos is using these for the construction permit application. We will have to V&V these codes and confirm the applicability of equations and correlations later.

MEMBER HALNON: Ben, I don't get that. I don't understand. I mean, that's like building something to a draft ASME code that's not approved yet and saying -- did they ask for approval of the codes and equations, or is this another one of those things at risk and they're just building moving forward, land unexplored?

MR. ADAMS: I believe it was made clear that all of these would be reviewed during the code V&V, which will be done at the OL stage. So we have not approved anything at this stage.

MEMBER HALNON: So that punch list is growing.

DR. SCHULTZ: Ben, this is Steve Schultz. Is there a schedule for that, for the submittals and your review, as well?

MR. ADAMS: We have not been given a specific schedule. We're assuming it's going to come in with the operating license application, but it could come in earlier. We don't know.

DR. SCHULTZ: In a topical report or something.

MR. ADAMS: It could come in a topical report, for example, yes.

MR. BEASLEY: So, yes, we do not have a schedule yet. We have talked with Kairos about some pre-application activities for the OL application, but we have not started those yet. I say we have talked with them. They brought it up. I think they are planning some pre-application activities, but, again, we have been focusing on this review and haven't started investing time in that yet.

I'm sorry. This is Ben Beasley with the NRC staff.

MEMBER HALNON: So to look back, I am trying to reconcile in my brain how you build something with unapproved codes and equations. How do you guys approve that you can go build this? Is it because of the hook and the operating license review? Is that what you're resting on from the standpoint of this thing could be built and just left there because you don't approve it in the future? I mean, is that essentially the strategy?

MR. BEASLEY: This is Ben Beasley with the staff. Yes, that is essentially the strategy. They have not asked for specific approval of any design or of these codes, and so, yes, they are accepting that risk.

MEMBER HALNON: Okay. I'll get off it. I don't fully understand the business model, but that's not our purview.

MEMBER REMPE: There were historical precedence years ago when they had the advanced reactor program and all the various types of designs. Some of them went all the way, some of them didn't, right. The PR, what is it called? The power reactor development program that was done back in the 50s and 60s, right.

MEMBER HALNON: That's kind of before I was in the nuclear business, but I --

MEMBER REMPE: It was before I was born. The history is interesting, though.

MR. ADAMS: Okay. I will be moving on to the NRC staff's technical evaluation conclusions. We'll go through the list of the PDC first.

So for PDC 10, reactor design, the Hermes thermal hydraulic design is designed to provide adequate heat removal. The NRC staff did perform scoping calculations, and the staff predicted that TRISO fuel does maintain integrity during postulated events and that there is adequate heat removal. This also played a role in the findings, and staff finds this is consistent with PDC 10.

For PDC 12, suppression of reactor power oscillations, we based our findings for the PDC on the fact that the FLiBe coolant should be experiencing single-based flow in the core and that the system has high thermal ownership. We did not have any specific analyses to review for the construction permit application, but, as stated in the PSAR, Kairos will be providing those with the operating license application for the inherent stability of the reactor. And the staff did not perform its own scoping calculations on this topic.

For PDC 34, residual heat removal, the PHTS residual heat removal during normal operations, which include start-up and shutdown, the downcomer fluidic diodes, reflector blocks, and THRS remove heat during postulated events and staff finds this is consistent with PDC 34. The PHTS DHRS will be discussed. I don't know if those are all later today, but I believe the DHRS is discussed later today. And the fluidic diode, we discussed that a little bit this morning.

For PDC 35, passive residual heat removal, again, the same features are relied on for heat removal during postulated events. All postulated events, the residual heat is removed via natural circulation through the downcomer and fluidic diodes and the reflector blocks and the DHRS. Staff finds this is consistent with PDC 35.

And, again, the scoping analyses performed by the staff played a role in performing our findings for all of these PDC. The information provided is consistent with PDC 10, 12, 34, and 35, and with acceptance criteria of NUREG-1537, and the analytical methods for the thermal hydraulic design are acceptable at this stage of the design.

The NRC staff concludes that the preliminary design provided in PSAR Section 4.6 is consistent with the applicable PDCs and acceptance criteria in NUREG-1537, and the NRC staff finds that the information in Hermes PSAR Section 4.6 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40, and further information can be reasonable left for the OL application.

I believe that is the last slide.

CHAIR PETTI: Members, questions? Let me just say this on our last discussion that I don't know that the risk is super high. There's huge margin in the design, so, you know, knowing what the number is, let's say, of a temperature, yes, there's uncertainty in that, but that the codes would be so wrong that you'd exceed something that would get you into trouble, I don't think that's the case. You just look at the power density and the heat removal capability of FLiBe and the conductivity of graphite. I think you can easily convince yourself that that's not an issue.

Okay. Then if there's no comments, we can take our break and, let's see, be back at 10:30.

(Whereupon, the above-entitled matter went off the record at 10:07 a.m. and resumed at 10:29 a.m.)

CHAIR PETTI: Kairos, are you ready?

DR. DORON: We're on. Okay. This is Oded Doron again, senior director of reactor system design. I'm going to be discussing PSAR Section 4.7, the reactor vessel support system, or what we call RVSS.

So the purpose of the RVSS, it supports the weight of the reactor vessel with the fuel, coolant, internals, and attachments. It provides thermal management to support the vessel expansion from ambient to full temp.

RVSS is a bottom support that includes a support tray, a ledge, support columns, support pads, base plate, vessel connector, and anchoring connectors. It is made out of 316H stainless steel. The reactor vessel bottom head sits directly on the bottom support. It's designed and fabricated using ASME Section III, Division 5. It's vertically anchored to the reactor building foundation.

RVSS thermal management. It protects the reactor building cavity concrete from thermal effects. Thermal break provided by insulation ensures the reactor building concrete, ensures the integrity of the reactor building concrete. Reactor building seismic isolation, it does not use lateral seismic restraints for the reactor vessel and head-mounted components the RVSS is designed to keep the reactor vessel from uplift and shear during seismic events. Design leverages seismic isolation of the reactor building to reduce seismic effects on the vessel, RVSS, and head-mounted components.

To the right there is a simple schematic showing where the RVSS will be located. Below that is some more rather light but still detailed on the design of the RVSS. I'll pause here for a second.

CHAIR PETTI: Yes, so just a question. The seismic isolator is below the RVSS?

DR. DORON: Well, rather, the reactor building is within, it's seismically isolated, so we take --

CHAIR PETTI: Oh, okay.

DR. DORON: Yes, we take credit for the input spectrum. Okay.

Now the PDCs. RVSS is designed to withstand the effects of natural phenomena and support the reactor vessel in the event of an earthquake. The bottom support and connectors meet ASCE 43-19 and preclude linear buckling of the vessel's forward columns and provide lateral and uplift support. This satisfies PDC 2.

RVSS is designed for the environmental conditions, including temperature loading cycles, in combination with mechanical loading cycles. Catch basin sensors for leak detectors are used to preclude damage to the RVSS from primary coolant leaks. It satisfies PDC 4.

RVSS design ensures the integrity of the reactor vessel during postulated events to support the geometry for passive removal of residual heat from the core by removing heat from the reactor vessel via the reactor thermal management system actively during normal operation and passively during postulated events. This satisfies PDC 74.

RVSS design removes heat from the vessel and ensures the integrity of the reactor vessel and the reflector blocks, thereby permitting sufficient insertion of the control and shutdown elements providing for reactor shutdown. RVSS design ensures that ACI 349-13 is met to support maintenance and inspection of the vessel bottom head and shell weld and reactor cavity. This satisfies PDC 74.

I believe this is my last slide.

CHAIR PETTI: Members, questions? Okay. Hearing none, let's move on to the staff then.

MR. LE: Good morning. My name is Tuan Le. I'm a reactor engineer with Division of Advanced Reactors and Non-Power Production and Utilization Facilities. Today, I will go over the staff review on the PSAR Section 4.7, the reactor vessel support systems. The agenda for this section I have -- next slide, please -- I will go over the PSAR Section 4.7, the reactor vessel support system; the regulatory basis; and PDCs. References the topical report, staff technical evaluation, technical conclusion, and regulatory findings.

Next slide, please. For our review of the reactor vessel support system, the RPD provide the structure support for the reactor vessel and reactor internals. Under these supports the full weight of the vessel, fuel, coolant, vessel internal, and head-mounted components. RDP is designed to handle the structures seismic and internal load to calculate the structures and address the thermal expansion during the initial heat-up and postulated events.

Next slide, please. The regulatory basis for this review of PSAR Section 4.7, the 10 CFR 50.34(a), the preliminary safety analysis report; 10 CFR 50.35, assurance of construction permits; 10 CFR 50.40, common standards; and the PDC associated with the review of this Section 4.7 are the following: PDC 2, design basis for protection against natural phenomena; PDC 4, environmental and dynamic effect design basis; the PDC 74, reactor vessel and reactor systems structure design basis. Our staff also used NUREG-1537, guidelines for repairing and reviewing the application for the licensing of non-power reactors.

For PDC 2, the staff evaluation as follows: PDC 2 is the design basis for protection against natural phenomena. Table 3 of 4.7-1, the load recommendation for reactor vessel supports provide a load recommendation for the RVSS include seismic loads due to the design basis earthquake. This load recommendation and the methodology will be used to demonstrate that the final design will meet the allowable stress limit specified in the ASME Section III, Division 5.

Based on acceptable loading methodology, the staff finds that the preliminary information for the RVSS design is consistent with PDC 2.

Next slide, please. For PDC 4, the staff evaluation as follows: PDC 4 is the environmental and dynamic effect design basis. The design features address the environmental and dynamic effect on the RVSS as following: pipe whip hazard included by design due to the lack of pressurized piping; discharge fluid addressed by catch basins with a leak protection system. The RVSS is designed to address temperature and mechanical load in order to prevent damage from creep fatigue to allow the thermal expansion of the reactor during the start-up and operations.

Based on design feature to manage the pipe whip hazard, discharge fluid, and the loading cycle, the staff find the preliminary information of the RVSS is consistent with the PDC 4 and the relevant NUREG-1537 criteria.

Next slide, please. For the PDC 74, staff evaluation as follow: PDC 4, reactor vessel reactor system structure design basis, RVSS design have to ensure there is sufficient heat removed from the vessel while also providing the structure support for the reactor vessel under static and dynamic load. This includes seismic loading. Heat is removed by the reactor thermal management system, the RTMS, actively during normal operation and passively during the postulated events.

Based on the design for sufficient heat removal and the structure support, the staff finds the preliminary information for the RVSS system design consistent with PDC 74.

Next slide, please. So in conclusion, the staff had the following findings: Regulatory findings, the staff finds that the preliminary design information is consistent with PDC 2, 4, and 74, and applicable criteria in the NUREG-1537. Staff concluded information in Hermes PSAR Section 4.7 is sufficient for the issuance of CP in accordance with

10 CFR 50.35, and further information can be reasonably left for the application stage.

That concludes my presentation for Section 4.3. I'll now open for any question, please.

CHAIR PETTI: Members, is there any questions?

MEMBER HALNON: We didn't hear a lot in your discussion about we'll look at that in the operating license portion. Previously, presenters have been very open with saying, hey, we need this but we'll get it in the operating. Is there anything that comes to mind that you think is very important? I know there's a lot that's going to be looked at in the operating, but what in your mind, are the important things to be looking at in the operating license when you review this 4.7?

MR. LE: We also identified some information detailing the design for the RVSS system, such as the loading cycle and profile for the RVSS system a detail like that would have to be reviewed to change the design for the systems. This information is supported for meeting the requirements, like stress analysis or support system, which includes testing the creed fatigue, the fatigue issue and the stress evaluation of the structures.

MEMBER HALNON: Okay. I know that the creep and the fatigue and those types of analysis, with only a four-year operating cycle, it's not necessarily a big concern, but I can understand that you'll want to see some of those cycles in the operating philosophies as we go forward.

Okay. Thank you.

CHAIR PETTI: Okay. Thank you. At this point, that concludes Chapter 4, so we can move to the memo. Thank you.

(Whereupon, the above-entitled matter went off the record at 10:44 a.m. and resumed at 11:00 a.m.)

CHAIR PETTI: Okay, since your slides are up, I assume you're ready, Kairos?

MR. PEEBLES: Yes, we're ready. This is Drew Peebles, senior manager of licensing. Nico Zweibaum is going to be presenting on the decay heat removal system, but I'm going to start us off on functional containment.

So, there's been a bit of discussion over the past two days on functional containment directly, or indirectly. So, this slide points out that the functional containment strategy is defined in commission policy, which is in SECY-18-0096. And that's a barrier, or set of barriers together, that effectively limit the physical transport of radioactive material to the environment.

There was some discussion on the reliability of the piping in the PHCS, and how we differ from LWR technology. So, there are a couple of parts of the SECY that I'll point out. One is that the idea of functional containment was born out of the differences from non-LWR technology from LWRs.

The LWR defense in depth strategy was based on the smaller margins for the zirconium clad fuel, and water coolant. So, advanced reactors having a more robust fuel design, a coolant that retains radionuclides, and operating conditions that don't allow for energetic releases when we have a break in the piping mean that we have a lot more margin to the consequences.

So, if I'm looking at Hermes specific consequences, think of it in terms of we have very large margins starting at the fuel. When we were talking about the head being leak tight, all of those we were kind of thinking of them in terms of physical barriers, again, beyond the fuel, and the salt. We don't need that, we don't rely on those, because even using the very deterministic safety case that we used for Hermes, we only credit the barriers in the fuel, the TRISO layers, and then the salt.

And we allow the radionuclides to transport to the site boundary, we still end up with consequences of one rem. So, this is significantly less than the 25 rem that's needed for a siting conclusion. But that is with the hyper conservative safety case where everything is -- we're not taking credit for it beyond the fuel, and the salt. So, I wanted to add that context before we moved on. Are there any thoughts, or questions on that first part?

CHAIR PETTI: You said the doses were one rem, or below one rem?

MR. PEEBLES: Below one rem.

CHAIR PETTI: Below, that's what I thought.

MR. PEEBLES: So, that's our target site boundary.

CHAIR PETTI: Right. I didn't see any hands, keep going.

MR. PEEBLES: All right, I'm going to hand it over to Nico for decay heat removal system.

MR. ZWEIBAUM: Thank you very much. As Drew mentioned before, my name is Nico Zweibaum, I'm the director of solid systems design here at Kairos

Power. So, as far as Chapter 6, and engineered safety features, the main system we described is our decay heat removal system. You see a picture on the left that kind of describes what that system looks like. Its purpose is to provide passive decay heat removal during postulated events when the primary heat transport system is unavailable.

The way this system is run, and works, it's an ex-vessel system. It continuously operates when the reactor is operating above the threshold power, and it removes energy from the vessel wall, thermal radiation, and conductive heat transfer to water-based annular thermosiphons, or thimbles. You can see on the picture, the vessel in the center, those annular thermosiphons are distributed around the periphery of the vessel.

And it's fed through water storage tanks, and steam separators. The DHRS, decay heat removal system, is shut off, and isolated when the reactor operates at low power levels. In that case the thermosiphons are dry, the parasitic losses alone are sufficient for decay heat removal. The DHRS is activated when the reactor starts operating above a defined threshold power.

And there is no change of state when that system is relied upon in response to postulated events. So, it's an always on type of system once we've crossed the determined threshold power at which we activate it. Of note, on the physics of that system, there's an inherent passive feedback mechanism, and the heat removal rate is a direct function of vessel temperature. And that's due to the physics of thermal radiation heat transfer.

So, there's a direct dependency between the amount of heat that we remove from the vessel wall, and the temperature difference between the vessel, and the thermosiphons where boil off is happening.

CHAIR PETTI: Just a question in terms of the activation of the DHRS, is it going to be automatic, or is it going to be an operator initiated action, or do you not know yet?

MR. ZWEIBAUM: That's something that we will clarify with the operating license application.

CHAIR PETTI: Okay, it's just you know that in the context of Part 53, there's these new definitions of types of facilities, if it's self-mitigating or not, and that all depends if the operator has a role in safety functions. So, it's just a point about the design, if we knew the answer, we could say something about it in the letter. That's fine, thank you.

MR. PEEBLES: So, this is Drew Peebles again, just to be clear, turning on DHRS is not a safety function. The rest of the mitigating functions, if there's a transient, would all be passive. And that is due to the fact that this activation happens before the system would be relied upon during a postulated event, right?

CHAIR PETTI: Okay, I understand, that helps, yes.

MEMBER BROWN: Can you say that again? This is Charlie Brown. Once you go into operation, or startup, and go critical, according to the chapter, it says it's not on until you pass a threshold, and at that threshold, you then have an automatic signal which turns it on, and it operates continuously when you're above that power threshold.

That seems pretty clear, in Chapter 7 it talks about an activation function which comes out of the RPS to turn it on, to ensure actuation of the DHRS. So, your statement that it's on all the time once you go above a certain power level is listed inside Chapter 6, which that makes it sound like it's part of the normal heat removal system.

MR. PEEBLES: I think we'd have to go back, and look at Chapter 6 again, but I don't think there's any RPS signal to turn.

MEMBER BROWN: Well, on Chapter 7.1, figure 7.1-1 says there's an activation function that goes from the RPS down to the DHRS system. And there's words in the chapter that say although the DHRS is always operating above a certain threshold of fission production accumulation level, the decay heat removal system provides actuation signal. The RPS provides actuation signal to DHRS to ensure it is operating when there is an RPS actuation.

And that goes on, and on. The actuation of the DHRS by removing the power to the water tank isolation valves to ensure passive flow.

MR. PEEBLES: Yeah, I'm being told it's actually a redundant feature. We may turn it on, or there's a safety signal. The safety signal would be in case it wasn't operating.

MEMBER BROWN: Well, let me back track. It says any time you go above that threshold power level of around ten megawatts, I'm sorry, that just flew out, I apologize, the DHRS is on all the time.

MR. PEEBLES: That is right, that is correct.

MEMBER BROWN: Well, that makes it sound like it's part of the normal cooling system for the system, for even normal operation.

MR. PEEBLES: Well, there will be some heat losses through that system during normal operations, yes. That's --

MEMBER BROWN: What if it fails during normal operation as you're at the maximum power?

MR. PEEBLES: So, we'll shut down anytime there is this type of anomaly. We'll have technical specifications on a number of metrics within that system, including water levels, and others, and integrity of the lead barrier. So, we'll get into some more details around the design of that system in a moment, there's a number of technical specifications that we'll be meeting with that system, and if we're outside of those bounds, then we would immediately shutdown the reactor.

MEMBER BROWN: Manually?

MR. PEEBLES: That would be defined in the tech specs of the operating license application.

MEMBER BROWN: In the Chapter 6 text, it says for half power operation greater than the threshold, it is in continuous operation, etcetera, etcetera. I'm just saying there's an inconsistency, in terms of my own mind, regardless, and you've got various other things like the tanks are capable of seven days operation, but yet there's other statements that it operates for 72 hours. I couldn't figure out the difference between those two times either. I'm sorry to be picky, it's just --

CHAIR PETTI: Let's let him get into that then, and if they address it, we'll go on.

MR. ZWEIBAUM: We can move on, but there will be mentions of the sizing of those tanks, and what it's being sized for, and that's the seven days. Next slide. So, this goes a little bit more into the various stages of operation, and hopefully sheds light on the high level statements I made on the previous slide. So, early on, let's say in the operating life of the reactor, when we're operating at low, or no power, you can see there that the isolation valves that's between those water storage tanks, and the thermosiphons, or thimbles, is closed.

So, the DHRS is deactivated. In that case, parasitic heat losses through the vessel wall, and cavity are sufficient to remove decay heat should a postulated event occur, and so we're not relying on decay heat removal from the DHRS in that situation to remove sufficient decay heat from the reactor. Next slide. So, once we operate above a defined threshold power, that isolation valve that was previously closed, opens.

We also have a feedwater line upstream of the water storage tank that is open. And so, we have continuous water flow from the feedwater system through the water storage tanks down to the thimbles, and continuous heat removal partly through that system. So, I think that's part of what Charlie was referring to before, which is that during normal operation here, we have some amount of parasitic losses through that DHRS system, and out through water boiling, and then steam evacuation through the atmosphere.

As far as the safety function of that system is really, if we go to the next slide, once the postulated event occurs, whether that be loss of the primary heat transport system, loss of electrical power, or loss of feedwater, in that case we're relying on the inventory of water that is in the water storage tank to continuously feedwater to the thimbles for up to seven days for decay heat removal from the reactor vessel.

MEMBER BROWN: Yes, it still sounds like it's required for operation for normal operation, put aside the postulated events. So, in other words your basic power operation does not remove enough heat to allow you to operate above a threshold, and DHRS is a critical system to make sure you cool the core. That's what it sounds like.

MR. ZWEIBAUM: I can take a little bit of that, I guess. So, the normal decay heat removal system, and heat removal system from that system during normal operations is through the primary heat transport system. So, the primary heat transport system is sized to extract all the heat from the reactor core during normal operations.

There happens to be some extra parasitic heat losses through the DHRS, which is by design, so that there's no change of state once that system is relied upon during onset of postulated event. But that is not strictly relied upon for heat removal during normal operations, if that's the question.

MEMBER BROWN: If you go above the threshold then, and you don't turn on the DHRS, the plant operates just fine with no problems?

MR. ZWEIBAUM: It would, but we will have technical specifications that would probably --

MEMBER BROWN: I'm not worried about the technical specs, I'm worried about the actual ability to operate now that you're no longer getting rid of these miscellaneous parasitic heat losses.

MR. ZWEIBAUM: It would technically be able to operate, the issue would be that on the onset of a postulated event in those conditions, if the DHRS were not on, that would mean that we were relying on a change of state, and that's what we're trying to avoid by design, which is why we turn it on before it's actually relied upon should a postulated event occur. But it is never relied upon for normal operations of the reactor.

MEMBER BROWN: So, it's got to be on already for normal operations in order to protect it from a postulated event?

MR. ZWEIBAUM: Yes.

MEMBER BROWN: Above a certain threshold, is that?

MR. ZWEIBAUM: That's right.

MEMBER BROWN: So, it's a safety system.

MR. ZWEIBAUM: Absolutely, that is our primary engineered safety feature.

MEMBER BROWN: Somebody said it's not a safety system a few minutes ago in the earlier conversation.

MR. ZWEIBAUM: It is.

MEMBER BROWN: There's four tanks, the way I read this, there's four sets of thimbles, four tanks?

MR. ZWEIBAUM: Yes, the answer is yes, but we'll get into some more details on the design, there's a few slides remaining, but yes to the statement you just made.

MEMBER BROWN: Why doesn't this then also require during postulated events, the feedwater is turned off to the tanks? Why do you do that? The valve is closed according to your picture. The previous picture, as well as the words in the text.

MR. ZWEIBAUM: So, that's something that we'll need to clarify for operating license depending on a number of failure modes analyses, but the main thing is that we have sufficient inventory in the tanks themselves so that we're not relying upon water coming from the feedwater system to feed those thimbles, and to remove sufficient decay heat over the seven day period that this is sized for.

MEMBER BROWN: What's the 72 hours that's mentioned in the text referring to them?

MR. ZWEIBAUM: I'm not sure which 72 hour text.

MEMBER BROWN: There was 72 hours

mentioned that I'm trying to find out. I can't have both my -- everything on, plus the slides at the same time.

MR. ZWEIBAUM: No, it's fine, I can elaborate on the timing. So, the seven days is meant for the storage capacity of the tanks to ensure there's enough water for the system to operate for seven days. The 72 hour period is advanced reactor -- excuse me, passive plant policy for safety-related systems to be able to operate. Beyond that, you can use non-safety-related systems to mitigate an event. In the case of the storage tanks, they are sized for the seven days' time.

MEMBER BROWN: Okay, so they're fundamentally sized for greater than 72 hours then?

MR. ZWEIBAUM: Correct.

MEMBER BROWN: Okay, so you more than meet the 72 hour requirement for this system. And the storage tank capacity does not need to be fed in order to meet that seven day requirement based on whatever level you establish in the tank as the normal level that you would be maintaining during normal operation?

MR. ZWEIBAUM: That's correct.

MEMBER BROWN: Okay.

MR. ZWEIBAUM: And the latest version of

the PSAR does clearly indicate the seven days, we'll make sure that this is, that there's no inconsistency in that document.

MEMBER BROWN: I guess my point being is there's a cycle of operation, it was somewhat piecemeal in the earlier version, or at least the version I had of the PSAR, and that's why I asked those. Also the activation based on the RPS, which is called out in both documents, the activation plus lock that's shown in the figure. So, that's not discussed in the text at all. That's a Chapter 7 thing, but it's also, I think it's mentioned in the -- the activation is not mentioned in Chapter 6, but it is mentioned in Chapter 7.

As a safety-related function, activating the DHRS, figure 7.1-1 shows it, and there's text that shows it also. I didn't write down the paragraph number.

MR. ZWEIBAUM: Yeah, so there is this mention in Chapter 7, that there is automatic activation by the reactor protection system.

MEMBER BROWN: So, it ought to be clarified what circumstances that's needed, because it wasn't clearly stated, at least I couldn't find it. All right, I got my points in, that's something to think about today, the interaction between Chapter 7 systems, and the DHRS should be further developed, that would be something I would include in your letter.

DR. BLEY: This is Dennis Bley, following up on Charlie's comment, if during operation, you do have some, I think you called it parasitic loss through the system, after a fair period of operation do you need to fill up these tanks again, or do they have enough water to cover that?

MR. ZWEIBAUM: So, during normal operation there is constant feeding of those tanks through the feedwater lines that you can see towards the right of this diagram. Yes, the water tanks are instantly replenished during normal operations. The goal of the capacity of those storage tanks is so that if a postulated event resulted in those feeder water lines not being available to replenish the tanks, then we have sufficient inventory to feedwater into the thimbles for up to seven days.

MEMBER BROWN: There was not a good discussion of that Dennis, of the level instrumentation. It was just noted, that's all. All right.

CHAIR PETTI: Keep going, thank you.

MR. ZWEIBAUM: So, a little bit more on kind of the process design for that system. One thing to note is that the DHRS does not directly interact with the reactor coolant. It's a system, it's a water based system, but it's completely outside of the solid boundary. As mentioned before, there's no change of state on onset of postulated events. The system is always on once we operate above set power levels.

There are several parallel, and independent cooling pathways. Four independent cooling trains, and only three trains are require to meet the cooling demands. So, I think there was a question before around there are four tanks, and four sets of thimbles, so this clarifies with only three be required for sufficient heat removal from the vessel.

MEMBER BROWN: I had another question, you talked about the parasitic thing is to keep something from going solid, is that the sodium?

MR. ZWEIBAUM: No, the parasitic losses are inherent to the fact that the system is activated during normal operations, there's no sodium in that system.

MEMBER BROWN: I know that they're not in that system, but you said keeping something from going solid in an earlier statement, and I didn't understand what you were trying to prevent from going solid, and if the DHRS was responsible for making sure that didn't happen during a postulated event, or something. I forgot the context.

MR. ZWEIBAUM: Sorry, the change of state statement might have been confusing. Change of state not in that it's liquid to solid, change of state in operating versus not operating.

MEMBER BROWN: Okay, not the condition of the coolant?

MR. ZWEIBAUM: No, no, no.

MEMBER BROWN: Okay, all right, thank you.

MR. ZWEIBAUM: The system is dual walled. You can see there's a leak barrier that encompasses all of the subsystems here, all the way up to above the water level in the water storage tank, and that is meant for leak prevention, and detection. But of note is that the system is designed so that we could continue to remove heat in the presence of a leak. And you see the isolation valve between the water storage tank, and the thimbles.

That's an active component, a failure would not introduce failures in heat removal, in that that valve fails in place. So, if the system operates, that is the valve is open, a failure would leave that valve open. And similarly, we have a float valve that regulates level in the separator above the thimbles, but that float valve stays open.

So, any failure of the system is designed to keep it removing heat from the vessel.

MEMBER BROWN: That's for the float valve, but the isolation valve, if it fails, it can fail closed.

MR. ZWEIBAUM: It fails in place. So, if it's open, it's all open.

MEMBER BROWN: Okay, thank you.

MR. ZWEIBAUM: Okay, so just diving down into a few of the subsystem designs. First the water storage tanks highlighted in blue on the right. The sizing, as mentioned a couple of times, is sufficient for up to seven days of continuous operation to support heat removal, and mitigate postulated events, even without replenishing through the feedwater line. The location of those tanks, four of them, they're outside of the reactor cavity.

That's the darker gray wall that you see to the left of the storage tanks. They're located at higher elevation than other DHRS components, and the flow from the water storage tanks into the thimbles is gravity driven, to the separator, and the thimbles.

As far as redundancy, and independence of those systems, only three out of four tanks are needed for adequate heat removal, and each tank is independent from the others in its location, and connection to the thimbles.

So, we have four fully independent trains of tank, separator, and six thimbles per train.

CHAIR PETTI: So, just a question, is this true even at, say above the threshold value? Because there's a threshold value, and then there's full power, that's quite a range of power. Is this all because the temperature on the outside of the vessel also scales with the power? So, everything kind of goes together.

MR. ZWEIBAUM: The temperature of the vessel will be maintained at the same value throughout the full operating range.

CHAIR PETTI: So, basically the radiant heat flux onto the thimble system is the same independent of what the power level is in the reactor?

MR. ZWEIBAUM: Yeah.

CHAIR PETTI: Then this is where I'm kind of mentally -- then there's a chance to overcool, because you can extract more heat than is in the core, because it --

MR. ZWEIBAUM: Yeah, so -- sorry, I'll let you finish.

MEMBER BROWN: It's turned off below a threshold.

CHAIR PETTI: I'm talking about above threshold.

MR. ZWEIBAUM: Yeah, so the system will be designed so that there is no over cooling, or freezing of the coolant inside the reactor vessel for up to 72 hours.

CHAIR PETTI: I understand that, I'm still confused. If you have a certain radiant heat flux, and that doesn't change, and there's a certain boil off rate of the coolant propositional to that radiant heat flux. Now, the power of the reactor could be above threshold, but the system doesn't know it, or doesn't feel it, is that?

MR. ZWEIBAUM: Yeah, that's correct. So, you'll see once the trend starts, the temperature evolution of the reactor vessel will depend on the initial power at which the postulated event happens. But the heat flux is constant, since it's just a dependency on the temperature difference between the reactor vessel wall, which is set at the onset of the transient, and the thimble wall temperature, which by virtue of boil off is set at 100 degrees Celsius.

CHAIR PETTI: Yeah, okay, so then temperatures inside the reactor are changing depending on where you are between threshold power --

MR. ZWEIBAUM: Yes.

CHAIR PETTI: Yeah, okay, it's clear now, thank you.

MEMBER HALNON: I may have blanked out on this, inventory for seven days, that's without feedwater makeup, correct?

CHAIR PETTI: Right.

MEMBER HALNON: Okay, I for some reason didn't get that.

MR. ZWEIBAUM: Okay, next slide. So, downstream of those tanks are the steam separators, and thimbles. The separator is pictured on the upper right, and also it is one of the figures in the PSAR. Those separators serve as the interface between the water storage tanks, and the thimbles. You can see that the feedwater line from the storage tank comes in on the upper right here.

There is a float valve that's located there to regulate level in the separator. So, when the water level exceeds a threshold value, that float valve blocks the feedwater line, and when the water level is below a threshold value, the float valve allows for continuous flow. It's a passive operation, and fail open design as mentioned earlier. If the valve fails, it fails open.

We will flood the separator, but we're designing, and analyzing the system to show that does not affect the net heat removal performance of the thimbles. The thimble itself is an annular thermosiphon that's located circumferentially around the outside of the reactor vessel, the four trains of six. We have a guide tube that's located at the center of that structure. That's the blue region that you can see in the picture on the right.

So, the water flow goes down through that guide tube, and then there's an ebb operator tube that's on the outside of that that forms an annulus where the boiling of the water happens for each removal. And yet a third annular region around that is the leak barrier that was mentioned earlier. So, we will have a dual well design that allows for the system to still remove heat, even if there's a breach in the evaporator tube.

And that's a fully passive flow system, so we're just relying on natural circulation, and boil off in those thermosiphons for heat removal.

MEMBER BROWN: So, the blue water comes out the bottom of the little blue pipe, and comes up around the outside of it as red, getting heated?

MR. ZWEIBAUM: That's correct.

MEMBER BROWN: Okay. And that gray area around it, I take it that's your thimble wall, and that's what's getting irradiated?

MR. ZWEIBAUM: Yes. But that's the leak barrier, so there is actually some void space in there, there's just gas in that region, and that acts as a leak barrier. So, if your red region leaks, you would still have water contained within that gray region, and it would not be flooding the reactor cavity. But we would detect that leak in that sense, and in the PSAR, and we'll have technical specifications also on potential shutdown of the reactor if we detect a leak in that region.

MEMBER BROWN: So, that's not a solid, okay.

MR. ZWEIBAUM: There's walls at each layer, but that gray region, and you can see -- it's very tiny font, I apologize for that, but it says gas at the very bottom right. So, that's a gas region outside of that.

MEMBER BROWN: That I don't understand.

The thimble is -- so, that gray is all gas?

MR. ZWEIBAUM: Yes.

MEMBER BROWN: Okay. So, you get radiated heat going into the outside wall thimble, then it's got a convective through the -- so, it gets through the gas?

MR. ZWEIBAUM: Yeah, you convect and radiate heat everywhere you have gas in that system out to the water.

MEMBER BROWN: But the gas is not a mobile type thing, it's a static system?

MR. ZWEIBAUM: Yeah.

MEMBER BROWN: Okay, thank you.

MEMBER KIRCHNER: Just a minor detail from a design standpoint, four thimbles like that would have to be either very large to get the surface area you need, or flattened out like a flat panel radiator. Have you done that part of the design? Because if you have localized small thimbles, then you're going to have local stresses in the vessel.

MR. ZWEIBAUM: Yeah, so we have run the analysis to show that the number of thimbles that we have around that system is consistent with the power level of the system. And so, for the 35 megawatt thermal reactor, the four trains of six thimbles, or actually, only three out of four are required, so technically 18 thimbles would be sufficient to remove sufficient decay heat during a postulated event to the maximum reactor power.

MEMBER BROWN: What's the diameter of a thimble? I mean are we talking inches?

MR. ZWEIBAUM: Yeah, it's a little under three inches.

MEMBER BROWN: So, you've got a total of 24 three-inch diameter thimbles going around the vessel?

MR. ZWEIBAUM: Yes.

MEMBER BROWN: So, is there a big gap between the thimbles then? I'm addressing Walt's question. They're not touching each other?

MR. ZWEIBAUM: They're not touching each other. So, I don't have the exact value of the gap off the top of my head, but they're equally distributed around the circumference of the vessel, and there's 24 of them.

MEMBER BROWN: And so they're sitting in air, or whatever --

MR. ZWEIBAUM: Yeah, they're sitting in the reactor cavity. So, the way the geometry goes, you have the vessel, you have those thimbles, and then outside of that you have the insulation that is sitting between the thimbles, and the reactor cavity wall, concrete wall.

MEMBER BROWN: Okay. Did that clarify your question? I had to ask that, Walt.

MEMBER KIRCHNER: It was a good question. Thank you.

MR. ZWEIBAUM: Okay, so going to the design basis for that system. First, the safety-related portions of the DHRS are designed to ASME 3 Division 5 Class B. ASCE 43-19, and 416 for seismic. And ACI 349-13 codes, and standards, and that's consistent with our principle design criteria one. That system is primarily located in the safety-related portion of the reactor building.

Which is designed to protect safety-related components from external hazards, and other non-safety-related DHRS components does not affect the components of the safety-related SSCs, that's PDC 2. That system is designed with low combustible materials, it uses physical separation of the trains to minimize the probability, and effect of fires, and explosions, PDC 3.

That system is designed with materials that will withstand the environmental conditions in the reactor cavity during normal operation, maintenance, testing, and postulated events. The components are designed to preclude cascading failures, and failures that could impact nearby safety systems, that's PDC 4. That system is capable of removing an adequate amount of decay heat to ensure that SARRDLs are not exceeded, and the reactor vessel, and fuel temperatures remain below their design limits, PDC 34, and 35.

And the system is designed to allow for periodic inspection, and functional testing to ensure integrity, operability, and performance of the system, and that's PDC 36, and 37.

CHAIR PETTI: Just another question, in terms of the temperature limits, in gas systems it's never the fuel that's the issue with your decay heat removal, it's always the vessel is closer to its limit, is that the case here too?

MR. ZWEIBAUM: Yes, so that system is designed to limit the vessel wall temperature to the maximum temperature level.

CHAIR PETTI: Right, okay, thank you.

MR. ZWEIBAUM: And its own, which is also specified in the PSAR. So, the metallic structures of the DHRS are also limited to that temperature value.

CHAIR PETTI: Yeah, right, okay.

MEMBER BROWN: Another question, this is Charlie again. If you're operating at full power, you're there for a while, long enough to be equilibrium, and the DHRS is on, and now you go fairly rapidly back down below the threshold, and yet below the threshold, the DHRS at least based on what's in the text, and everything, would turn off.

MR. ZWEIBAUM: Not necessarily.

MEMBER BROWN: Well, that's the implication, I'm not saying -- just that was the implication, because it comes on when you go above the threshold automatically. That's at least the implication from reading the text in Chapter 6, and Chapter 7. And if you go back down below that, and it goes off, isn't there a greater amount of parasitic, or whatever other kind of heat you want if you reduce it fairly rapidly back down?

Like you shutdown immediately, and now the DHRS goes off when you go below the threshold power, has that been considered?

MR. ZWEIBAUM: Yeah, so at that stage the operations of the DHRS, and turning on, and turning off will be a function of the power history of the reactor, and action related fission products, and the decay heat levels that result from that as opposed to a straight power level. So, it's more of a power history dependency at that stage compared to just a straight on, off depending on power.

MEMBER BROWN: Well, you've got to have some instruments that determine what that is, and then make sure it stays on if you come back down, and stays on for a while before you turn it off, wouldn't that be correct?

MR. ZWEIBAUM: Yes. We'll have neutron flux detectors, and a number of other metrics that will allow us to determine that.

MEMBER BROWN: Well, it's the operation, the time at power that builds up the decay heat that you've got to deal with. If you suddenly go down, if you shutdown, now all of a sudden you've got to deal with that. And that's higher than what you have from sitting around for a while as you're going up and go over the threshold. There's some complications in there, and something's got to measure it and turn it on and off. That's all I'm saying.

I haven't completed reviewing all of Chapter 7 yet, so I focused kind of on the decay heat removal system part of it when we were doing this review.

MR. ZWEIBAUM: With --

MEMBER BROWN: And with the activation -- I'm sorry, I interrupted you, sorry.

MR. ZWEIBAUM: We'll clarify all of this with the operating license application. There will be a number of different scenarios that we'll have to consider for this turning on, and turning off that will be more detailed at that stage. Those points are all valid.

MEMBER BROWN: Will that also be -- because very clearly it looks like the RPS is utilized right now for the turning on, or the activation, presumably that would be for deactivation as well, and that would be the system that would have the information. It would seem the algorithm is necessary to determine what the power history is, and what you might need to do.

Nothing says you can't do that with the computer based systems, and the algorithms you can use. It's just got to be addressed, that's all.

MR. ZWEIBAUM: So, we did bring our director of instrumentation, and control to sort of clarify the message earlier about actuation using RPS.

DR. CILLIERS: Great, thank you very much, this is Anthonie Cilliers speaking, director of instrumentation, and controls. So, I think we'll discuss a little bit more during Chapter 7, but the RPS does have a function in the activation of the DHRS system. How it works, and it's described in Chapter 7 a little bit more detailed, is that the DHRS activation, and deactivation is in full manual control.

So, the operators have full control of that system before that system is required for its safety function. In other words, before you have enough accumulated fission products in the system. At some point, and we will have to clarify that at the ORA stage, what that exact algorithm would look like. There will be a determination based on instrumentation, power level, flux level, temperature.

There will be a determination shortly before the DHRS becomes a safety function where RPS will activate the DHRS system, and remove manual controls from the operators. In other words, the operator will in that case, be unable to deactivate DHRS because of that. Once an event occurs after that, and we could include if an event occurs before DHRS is activated, or after DHRS is activated.

The RPS will monitor the system, and that is completely temperature based. When the temperature has been reduced sufficiently, that there is no risk to any of the safety functions that the RPS is protecting, it will hand manual control back to the operators, so that the operators can choose to activate DHRS. Because that is not a safety function, the deactivation. I hope that clarifies it.

MEMBER BROWN: Yeah, I was the hand off, I read the part about the hand off in Chapter 7 so I would get a little bit of an understanding of how it interacted with the DHRS. But it wasn't as detailed as your explanation, so it's just we need to have that clarified when we get down to the operating level, operating license time of this, to how all that stuff interacts.

And you only show four channels, but you don't show any details on what the RPS looks like. But that's for later, you don't need to do that now, we just need to know you're going to do it. So, all right, thank you, that helped, I appreciate it. I didn't have anything else Dave.

CHAIR PETTI: Yeah, anybody else? If not, we'll turn to the staff.

MS. SIWY: Hi everyone, I hope you're all doing well today. My name is Alex Siwy, and I'm a technical reviewer in the division of advanced reactors, and non-power production, and utilization facilities. I will be presenting a staff's review of Chapter 6 of the PSAR, engineered safety features. As you will notice, we start with section 6.2, because section 6.1 is a summary description, and there are no specific safety findings to be made.

Next slide, please. So, here's an outline of what I will be discussing. First, a very brief overview of PSAR section 6.2, functional containment. And I'll provide the regulatory basis, as well as the staff's technical evaluation, and technical conclusions, and wrap up with regulatory findings. Next slide, please. So, I think Kairos did a good job of covering this already.

But just to reiterate, the Hermes design uses a functional containment to limit the potential release of radioactive material. The functional containment includes physical barriers, operating conditions, coolant design, and fuel form. So, for example the TRISO fuel retains radionuclides, and the layers of the TRISO particles form barriers. The pebble itself does provide some physical protection to the TRISO particles.

In addition, Hermes will be operated such that a large margin to the TRISO fuel design temperature is expected. The FLiBe coolant is credited for retaining radionuclides that are not aerosolized, or evaporated during an event. And Hermes operates at a near atmospheric primary system pressure, which limits the driving force for radionuclides to reach the environment in the case of an event.

It's also important to note that PSAR section 6.2 describes the overall strategy of using a functional containment, but Chapter 13 is really where the implementation, and the demonstration of its acceptability is covered. Next slide, please. This slide covers regulatory basis. The 10 CFR regulations are very familiar by now. 50.34A, 50.35, and 50.40. The one unique aspect for the functional containment is the SECY paper, SECY 180096, functional containment performance criteria for non-light water reactors.

And it's approval in the associated staff requirements memorandum. Next slide, please. So, to give you an overview of how the staff conducted its review, the staff used relevant parts of NUREG-1537 as guidance, and in particular the sections included confinement, and containment, since the high level objectives are basically the same as a functional containment.

We also ensured that the functional containment approach was consistent with the functional containment SECY. The one difference we noted was that the Hermes design uses a maximum hypothetical accident approach consistent with NUREG-1537, as opposed to the event category approach that most power reactors use. Finally, the NRC staff ensured that the staff was evaluating each individual component, and feature of the functional containment, as well as its integral performance.

And those PSAR sections, and the corresponding staff safety evaluations are in section 4.2.1 reactor fuel, 4.3 reactor vessel system, 5.1 primary heat transport system, and 13.1.1, and 13.2.1 which cover the maximum hypothetical accident. Next slide, please. So, this slide just summarizes the conclusions that we made relative to NUREG-1537. PSAR Chapter 13 demonstrates the need for functional containment as an engineered safety feature because it is credited to mitigate the Chapter 13 events.

The preliminary MHA analysis in Chapter 13 suggests that the radiological consequences are within the 10 CFR Part 100 criteria, but that's something that we will be confirming as part of the operating license application review. In addition, the functional containment has no adverse impacts to normal operations, or safe shutdown, as it is an inherent part of the design.

There will be safety-related instrumentation to monitor the components of the functional containment. And finally, the technical specifications will confirm continued operability of the functional containment, and they are important to ensure that actual dose consequences will remain bounded by the MHA analysis. Next slide.

CHAIR PETTI: Just a question back on the instrumentation, you're thinking of like level of salt, or?

MS. SIWY: Yeah, level in the salt, temperature indications for fuel, things of that nature.

CHAIR PETTI: You can't measure the temperature of the fuel easily, but salt.

MS. SIWY: Yeah, yes. And to summarize the regulatory findings, the NRC staff finds that the preliminary design information is consistent with the applicable criteria in NUREG-1537. The functional containment approach is consistent with the functional containment SECY, and staff requirements memorandum, and the staff concludes that the information in Hermes

PSAR section 6.2 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35, and 50.40.

And further information can reasonably be left for the operating license application.

CHAIR PETTI: Just a question, were there any scoping calculations done of the DHRS, like other parts of the plant we heard yesterday, that look at heat removal, and things?

MS. SIWY: That was included as part of some --

MR. SCHMIDT: Yeah, this is Jeff Schmidt of record systems, so you're going to see that in Chapter 13. Yeah, it was modeled as part of the MELCOR.

CHAIR PETTI: Okay, great, thank you. Questions members?

MEMBER KIRCHNER: Dave, this is Walt.

CHAIR PETTI: Yeah.

MEMBER KIRCHNER: Functional containment is not new, I mean the whole TRISO fuel program is -- that's well understood. My question to the staff is what are you looking for in terms of the credit that's being taken for the salt retaining radio nuclides in terms of evaporation, release, unmitigated air interaction with the coolant, etcetera. Is that part of the Appendix A set of programs that have been identified?

MS. SIWY: So, that's not specifically my area, I apologize. But I do note that -- or it's my understanding that there is no credit taken for anything that aerosolizes, or evaporates out of the FLiBe, it's only what remains within the FLiBe, I don't know if that answers your question.

MEMBER KIRCHNER: Yeah, that's exactly the question. So, is there an understanding of the chemistry of the FLiBe, is that sufficient to lead you to the conclusion that even if you have a spill, or the primary system is broached, and it's open to air, that the FLiBe will retain the radionuclides there in solution, and carried by the salt?

CHAIR PETTI: So, Walt, just if you recall, in our letter, we reviewed this in the source term topical, and we actually made a plan about having some data to back up the assumptions, and the staff accepted that. And I believe, then, it is in Appendix A.

MEMBER KIRCHNER: Yeah, that was my understanding Dave, I just wanted to clarify.

CHAIR PETTI: Yeah, it's a little

different wording, but is that -- it's in Appendix A, right?

MR. SCHMIDT: That's what we were just discussing. I'm not sure it's in Appendix A, but it's part of the mechanistic --

CHAIR PETTI: So, something about justification of thermodynamic data, and that's what I interpreted that to mean. It's in the R&D, I think.

MR. SCHMIDT: All right, so I guess the answer is yes.

CHAIR PETTI: Okay, thank you. Okay, so members, if you'll indulge me, first of all, the letter was written by Jose, but he's not here, so I told him I would fill in for him. I don't think the letter will take that long, we're eight minutes from -- we have 6.3? Sorry.

MEMBER KIRCHNER: Don't we have regulatory findings on the DHRS?

MS. SIWY: We can just keep going.

CHAIR PETTI: No, keep going, sorry, keep going. Okay, I think we can hop a couple slides ahead Ed. I think we're missing 6.3, it should be -- okay, there we go. So, I'll also be presenting the staff's review of the decay heat removal system. Next slide, please. Same agenda as last time, except decay heat removal system. Next slide. So, a very brief overview of PSAR section 6.3.

Again, the DHRS is a safety grade system that's credited to remove decay heat when normal heat removal is not available, such that reactor vessel temperature will be maintained within the acceptable limits for 316 stainless steel. The DHRS is designed to provide passive cooling for up to seven days following a postulated event without electrical power, or operator action.

Again, the main components are the water storage tanks, the steam separators, and the annular thermosiphons. And the mechanism by which it works is that the heat from the reactor vessel is transferred via radiation, and convection to water in the annular thermosiphons where the water boils off. And we discussed in great length how DHRS is placed in service of the threshold power, where the passive radiative cooling alone is not adequate. Next slide, please.

MEMBER HALNON: Alex, on the passive, the failed in position for the valve is very important, will you be looking at power supplies, if it's a DC valve, what a potential short could do to change state, and that sort of thing? I assume that's not the detail that we have now. But that failure mechanism is something we had to deal with in light water reactors, the smart short issue, those types of things. Those are all part of the review criteria?

MS. SIWY: Yes, we're looking very heavily at various failure modes.

MEMBER HALNON: Okay, thank you.

MS. SIWY: Thank you. So, the regulatory basis for the DHRS is the same 10 CFR Part 50 regulations, and several NRC staff approved PDC from the Paris power topical report for PDC. They're listed on the slide, I won't go through them. Next slide, please. So, to give you a sense of the review process for this section, we used relevant parts of NUREG-1537 as guidance.

In this case there's nothing quite like a passive DHRS in there, so we looked at the emergency core cooling system, and secondary cooling, and kind of took the bits, and pieces that were applicable from those systems, as well as the overall guidance on engineered safety features. We also reviewed the DHRS against the PDCs that were listed on the previous slide.

As I mentioned just a little bit ago, the staff reviewed the preliminary system design to identify potential system failure modes. We also audited the Kairos detailed system description, and the DHRS heat performance calculation that determine the level of system performance needed to maintain the vessel temperature below the limit for 316 stainless steel.

And, finally, we performed independent calculations to ensure that the water tank volumes would be sufficient for seven days' worth of cooling. Next slide, please. A couple of aspects of this review that I wanted to particularly highlight for the Hermes design is again going back to our focus on identifying important phenomena and potential failure modes.

We ensured that the Kairos testing plans address those types of things, and examples include potential flow, and heat removal, and stability issues during the transition, and in service phases. Potential dynamic loads on structure, and components due to transition phase thermal shock, and in service evaporative boiling, and the potential for corrosion, and fouling within the evaporator tube.

The other really unique aspect of this review is that the DHRS needs to be able to both accommodate the highest heat loads for maintaining vessel integrity, as well as the lowest in service loads to prevent freezing per a requirement in Chapter 13 of the PSAR. And this is something that we have noted, we will definitely be looking at how the design meets these competing design requirements as part of the OL application review.

Next slide, please. And to summarize the evaluation against the principle design criteria, PDC 1 requires safety significant SSCs to be designed, fabricated, erected, and tested to appropriate quality standards, and identification of generally recognized codes, and standards used. As was mentioned, the DHRS will be designed to several codes, and standards, ASME, ASCE, ACI standards.

The component design to these standards is evaluated in Chapters 3 and 4 of the staff safety evaluation, and the quality assurance program is evaluated in Chapter 12 of the safety evaluation. PDC 2 requires protection against natural phenomena. And the way this is addressed with DHRS is that it's located in the safety-related portion of the reactor building, except for the steam vent lines, and the failure of the steam vent lines would not impede the safety function of the system.

And the staff evaluation of seismic

methodologies are in SCR sections 3.4 and 3.5. Next slide, please. PDC 3 requires protection against fires, and the DHRS will be designed with low combustible materials and physical separation. The overall fire protection program is evaluated in SER section 9.4.

PDC 4 requires protection against environmental and dynamic effects, and the DHRS will be designed to withstand normal operating conditions, and to protect against dynamic environmental effects.

Finally, PDCs 10, 34, and 35 address adequate heat removal under normal operation and postulated events. And as was noted, the DHRS is designed with redundancy to protect against single failures, both with the redundant train, and there's actually a redundant thimble in each train. And preliminary calculations within the PSAR suggest that the DHRS does remove heat adequately.

Finally PDCs 36 and 37 address inspection, and functional testing of the DHRS. The DHRS is designed for online monitoring, and access to perform inspections, and it will be functionally tested during the startup phase.

Next slide, please. So, here are the conclusions relative to NUREG-1537. PSAR Chapter 13 demonstrates the need for DHRS as an engineered safety feature through it being credited to mitigate events. The preliminary PSAR analysis suggests that the DHRS does remove adequate amounts of decay heat, and that the radiological consequences are within the 10 CFR Part 100 criteria. And again, staff will confirm this as part of the operating license application review.

The DHRS has no adverse impacts to normal operations, or safe shutdown. In fact is part of normal operation above the threshold power level. The technical specifications, as well as normal operation itself will help to confirm its continued operability and availability when needed. And one thing that I really want to drive home is that the DHR test program will be key to finalizing the DHRS design.

Next slide, please. And to summarize the regulatory findings, the NRC staff concludes that the preliminary design is consistent with the associated PDCs, and acceptance criteria in NUREG-1537. And the NRC staff finds that the information in Hermes PSAR Section 6.3 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40, and that further information can reasonably be left for the operating license application.

Are there any other questions?

MEMBER REMPE: So, I have a question about the exchange before you started this part of your presentation. When Walt asked the question, and the response was well that was actually something we brought up in the topical report on the source term, and we had a recommendation, was that accepted -- that isn't in Appendix A.

How does that get communicated with the applicant that not only what's in Appendix A, but also that isn't the only topical report we had recommendations on where the staff agreed that yes, something else is needed as I recall. And I'm just wondering is all of that going to be tracked, and how easy it for everybody to review it?

MS. SIWY: So, as part of the staff's review of each of the sections that utilized the methodology, the staff should be checking the limitations, and conditions, and recommendations associated with the topical report safety evaluation.

MEMBER REMPE: Because, again, hopefully ACRS will also remember, and the applicant is well aware that everyone -- there's no confusion.

CHAIR PETTI: But everyone mentions the R&D items, one of the R&D items is justification of the thermodynamic data, etcetera, etcetera. That's the response to our concern on the source term.

MEMBER REMPE: Okay, because the response back, the actual recommendation, and unfortunately my computer is doing weird things, but it was about the vaporization, and all that. And --

CHAIR PETTI: Yes, we asked that they have, that they get data behind that model.

MEMBER REMPE: And that'll be clear to the applicant and the staff that we're talking about that. I'm just wondering if it needs some expansion. But if they checked what's in the topical report reviews, that's good. But having that appear in Chapter 1, I kind of sensed there was some disconnect at first, until Dave mentioned it, and that's why I'm asking the question. I want to make sure that there won't be a disconnect two or three years down the road.

MR. BEASLEY: Yeah, this is Ben Beasley with the staff. So, to just re-emphasize what Alex said, so when we reviewed the PSAR, we reviewed that in conjunction with the topical reports, and the conditions that were in there, and we'll do the same with the operating license application.

MEMBER REMPE: The conditions plus the ACRS response to our letters that you agreed with what we suggested.

MR. BEASLEY: Right, well, so if we agreed with your suggestion, that should have been included in our safety evaluation, so that will be in the dash A version of the topical report.

MEMBER REMPE: Good, okay, just wanted to make sure, thank you.

MR. BEASLEY: Yes.

CHAIR PETTI: Other questions, members?

MEMBER KIRCHNER: Dave, may I make an observation?

CHAIR PETTI: Sure.

MEMBER KIRCHNER: And it ties back to the earlier discussions about the core, the reactor design. I would just suggest this to both the applicant, and the staff. It would not be that hard to make a bounding calculation on the thermal performance of this reactor system with the decay heat removal system functioning to see what the core center line peak temperatures are in the TRISO particles in a decay heat cool down situation without the fluidic valve functioning.

In other words, just assume you have a static system, ignore convective heat transfer so to speak, and in the down comer, natural circulation, and just calculate what the core peak center line temperatures are for the TRISO particles. I would hazard a guess that these power levels, since you probably will not get to equilibrium with this core, and this is speculative on my part, that the decay heat removal system functioning will keep the core below the threshold for significant damage in the TRISO particles.

But it would be interesting to see, because then if you have this calculation as a function of power, or more correctly, decay heat load, then you could set a bar for startup testing after the operating license in terms of proving out the fluidic diode in particular, and its function. Just an observation, thank you Dave. And it's not that hard a calculation to do.

You make some conservative assumptions about the radial conduction in the pebble bed, the reflector in the down comer, and from the vessel to the decay heat removal system. My sense is that with the powers that you'll see, and the radius of this vessel, that you would keep the peak temperatures of the TRISO particles well below their limits.

CHAIR PETTI: That's my sense, too, having seen many of these sorts of calculations in the past.

Okay, so now we're done. Sorry. So, I

know we're beyond our time, but it would be nice to be able to finish this memo, and then break for lunch, I think. We'll be ahead of schedule. We'll only have one chapter after lunch, which will be, I think, good.

(Whereupon, the above-entitled matter went off the record at 12:07 p.m. and resumed at 1:14 p.m.)

CHAIR PETTI: Are you ready, Kairos?

MR. ZWEIBAUM: Yes. Ready and unmuted now.

CHAIR PETTI: Okay. Whenever you want to start.

MR. ZWEIBAUM: Okay. So I'll kick us off on this chapter. This is Nico Zweibaum, director of salt systems design. This is Chapter 9 on auxiliary systems. And there is quite a variety of those systems, so you'll see a corresponding variety of presenters throughout.

So as I said, I'll kick us off. Next slide, please. So, as an overview for reactor auxiliary systems, the reactor coolant auxiliary systems are a collection of systems that provide support for the functionality and performance of FLiBe, our reactor coolant, to remove fission products, activation products, and other chemical impurities and particulates from the reactor coolant to maintain the cover gas atmosphere, both in pressure and composition in the head space above the core, to provide removal and storage of tritium, to control inventory, fueling and draining processes for systems that contain reactor coolant, including transfer of coolants into the reactor at the beginning of life, and to provide active and passive thermal managements to reactor system components.

So these various functions are implemented into the following reactor coolant auxiliary systems. We have a chemistry control system, inert gas system, tritium management system, inventory management system, that's the salt inventory, and reactor thermal management system.

Of note, and this will be a theme throughout this presentation, but those systems are not accredited with performing any safety-related functions.

So, starting with the chemistry control system, or CCS. Again, not created with performing any safety-related functions, but what the system does is it monitors primary coolant chemistry for compliance with FLiBe specifications. The system extract coolant samples for an offline analysis for the FLiBe chemistry.

As far as design basis, due to the proximity of the CCS to the reactor vessel, it's designed so that seismic induced failure does not impact the reactor vessel system, it's consistent with PDC 2. Again, due to the proximity with the reactor vessel, that system is designed so that adverse effects of postulated failures do not impact the reactor vessel system, consistent with PDC 4.

The CCS will monitor the reactor coolant purity with offline sampling analysis to determine if the reactor coolant is within specified design minutes, consistent with PDC 70. And consistent with 10 CFR 20.1406, the CCS is designed, to the extent practicable, to minimize contamination of the facility and the environment and facilitate eventual decommission.

MEMBER HALNON: So that last statement is pretty much a theme throughout this whole chapter. What does that look like?

I mean, are you going to have a process, some kind of board or some kind of hold point in your design review that sits back and asks the questions, the hard questions, whether or not this could be done a different way?

MR. ZWEIBAUM: Well, we'll clarify that by the operating license application.

MEMBER HALNON: I get that. So you haven't thought through it yet, is that the case?

The reason is, is that that, I mean, that statement is pretty prominent in this chapter, more than the others. And if you're just putting it to the OL, I get that.

MR. GARDNER: So this is Darrell Gardner from licensing. So, I mean, obviously it's a rule we have to satisfy. There is a number of ways to do that, depending on the system, terms and materials you use, maybe layout of things like that.

There is, you know, you could look back at some other design certification FSARs and look at sort of the programmatic descriptions that they have included about how they are going to do with the satisfying 1406. And I would suspect that we would do something similar.

MEMBER HALNON: Okay. Yes, that makes sense, Darrell. It seems like there would be a system approach to generically looking at a system and equipment and either using the same type of sets of questions or sets of challenges. Having a challenge, or something to that effect. It just seems like it's right for a process that might be consistent across the board.

MR. GARDNER: Sure is.

MEMBER HALNON: All right, thanks.

MEMBER BALLINGER: Now this is Ron Ballinger. You know that the first five bullets are inconsistent with the last one, in the sense that this all argues for online monitoring. Now I'm wondering whether you've thought about that?

CHAIR PETTI: Oh, you mean the Part 20 requirement could push you to online monitoring.

MEMBER BALLINGER: Well online monitoring period. Everything.

CHAIR PETTI: But that's more complex in taking a sample.

MEMBER BALLINGER: Okay.

CHAIR PETTI: In principle.

MEMBER BALLINGER: Remember, it's hot salt.

CHAIR PETTI: I know it's hot salt.

MEMBER BALLINGER: Okay.

CHAIR PETTI: I understand that. Yes.

MEMBER BALLINGER: And so, from the standpoint of physical safety and those kinds of things, not having to deal with hot salt, radioactive hot salt complication. Anyway. I mean, what parameters are you measuring?

MR. ZWEIBAUM: So the system's function is to make sure that the FLiBe is consistent with the specifications that were set forth in Topical Report 05, which is our reactor coolant topical report.

MEMBER BALLINGER: Right. So, in theory, if you had a good calibration, all you need to do is measure the electrical, the chemical potential.

MR. ZWEIBAUM: Yes. So how we will do this will be specified, but this is essentially the function this is serving. Yes.

MEMBER BALLINGER: Okay. I just, you know, you've got an electrode in there already, right?

MR. ZWEIBAUM: Yes. There will be a number of systems that are part of this chemistry control system that will be interfacing with our inventory management system, as I'll mention later, so we're taking regular representative samples from the salt and comparing to the tech spec.

MEMBER BALLINGER: Okay, thanks. It's just that the BWR folks have gone, went from grab samples to online monitoring. And I think the BWR people did, and so did the Navy people.

MR. ZWEIBAUM: We'll clarify that further by OL. Okay.

So, separate system. The inert gas system. Again, that system is not credited with performing any safety-related function. Its role is to provide inert argon gas as a purge flow to system components during normal operation and maintenance. Part of that system will be removing impurities from the cover gas. The system will also provide reactor coolant motive pressure during filling and draining operations.

As far as the design basis, similar to the previous system, this IGS may be in proximity or connected to safety-related SSCs and across the seismic isolation mode. It's designed so that seismic induced failure will not impact safety-related SSCs from performing their safety function, consisted with PDC 2.

It is a low pressure system and precludes pipe whip. Nearby safety-related SSCs will not be affected by escaping inert argon gas consistent with PDC 4. That system will monitor radioactive levels in the gas to support the evaluation of the radioactive material releases that might occur as a result of a system or fuel failure, consistent with PDC 64. And then you got the same statement as before.

MEMBER HALNON: Yes. My favorite 20.1406.

Hey, my notes, when I reviewed this, said it also was meant to operate during postulated events. That's left off this slide. Was that intentional or is it a change, or what's the case there?

MR. ZWEIBAUM: It's not credited to performing safety function during postulated events.

MEMBER HALNON: Okay. But it is there and it does --

MR. ZWEIBAUM: Yes.

MEMBER HALNON: And it does measure, at least perform as a cover gas. And so there is no confinement activity, or not activity, but functions done by this gas then?

MR. ZWEIBAUM: No.

MEMBER HALNON: No retention and credits taken. Okay. But my sense is, is that it probably should be on this slide so we could have had that discussion. So go on. Thank you.

MR. ZWEIBAUM: Yes.

MR. DOLAN: All right. So, this is Kieran Dolan. I'm the responsible engineer for the tritium management system at Kairos. I'll be presenting this slide.

So tritium is produced by neutron irradiation of the FLiBe coolant. Tritium management system, or TMS, captures some of that tritium to prevent environmental release. And it does that with the tritium capture subsystem integrated into the inert gas system. And another tritium capture subsystem integrated into the reactor building HVAC.

So the tritium management system is not credited with performing any safety functions. Like I said, it separates tritium from argon and the inert gas system, and from dry air and reactor building cells.

The tritium management system collects and temporarily stores tritium to facilitate final disposition in accordance with 10 CFR 71.51. Type A and Type B packing canisters are used to package and transport tritium, depending on activity levels.

Due to the potential proximity of the TMS to the reactor vessel, TMS is designed so that seismic induced failure does not impact the reactor vessel system, consistent with PDC 2. Tritium monitoring sensors are selected and provided over an approach range of tritium activities where the measurements are needed, consistent with PDC 13.

Tritium management system captures tritium at an overall efficiency to minimize tritium releases to the plant in accordance with PDC 60. And radiation monitoring is provided in the TMS for the evaluation of tritium levels in TDS subsystems in support of evaluation of radioactive material releases that might occur as a result of system failure consistent with PDC 64.

And we also have the statement here consistent with 10 CFR 20.1406, the TMS is designed to the extent practicable to minimize contamination of the facility and environment and facilitate for decommission.

CHAIR PETTI: Question. What technology are you going to use to separate the tritium from the argon in the air?

MR. DOLAN: So described in PSAR, we're using the technology of metallic getters. So specifically, in the system design description for the tritium management system we were looking at zirconium based getters. So zirconium iron, trade name ST-198 from SAES getters. That's the baseline as of the PSAR.

CHAIR PETTI: Thank you. Could you provide a little more detail on the used, the used beds, used capture beds being stored or put in unqualified storage canisters? I may not have phrased that correctly, but I remember a statement in there saying that they were initially put into unqualified canisters. Not qualified for storage, for tritium.

MR. DOLAN: Right. So the canisters we're using are shipping canisters, not storage canisters. So they will be in the plant for a period of time. But I think the intent of our statement there was to clarify that the canisters are not for long-term storage, they're really just transportation canisters to get the used beds to a low-level waste disposal service.

CHAIR PETTI: Okay, thanks.

MEMBER BALLINGER: I have a question, which I'm sure the answer might be obvious, but with respect to the Handbook 69 values, or whatever the release limits are for tritium, how close are you to being able to dump this stuff up a stack?

MR. DENMAN: So this is Matthew Denman. The radiological source term responsible engineer. Can I just clarify, your question is on steady state effluents, right? How close are we to being able to not hold up any tritium and release the tritium out the effluents stack?

MEMBER BALLINGER: Yes.

MR. DENMAN: In our Chapter 11 analysis, that's effectively what we did. We did not credit tritium hold up in the tritium management system. The entire source, or generation rate for the tritium was released through the effluent stack as a band founding calculation. And we were still within the Part 20 limits.

MEMBER BALLINGER: Wow, okay. Thanks.

CHAIR PETTI: Matthew, since we're there, I thought we'd probably hit this in 11. You modeled it as HT, and left the dose calculation converted to HTL?

MR. DENMAN: So we assumed that as soon as it permeated through the heat rejection radiator it would immediately oxidize and then transport as HTL.

CHAIR PETTI: Okay. And so, I can understand meeting sort of the offsite dose limit, but eventually that tritium gets into drinking water. Are you saying that you meet the drinking water standard as well?

MR. DENMAN: The drinking, so we evaluated against the Part 20 dose limits. We did not evaluate against any other regulatory limits.

CHAIR PETTI: Yeah. Well, this is always an issue with tritium, is that that stuff eventually gets into the groundwater and drinking water. And that limit is really, really tight. So it might be worth looking at that.

MR. DENMAN: Yes.

CHAIR PETTI: Because it's an EPA limit of, what is it, Part 40 I think.

MR. DENMAN: Thank you for your comment.

CHAIR PETTI: 20,000 picocuries. Yes.

MR. DENMAN: Understand. And thank you for your comment.

MR. ZWEIBAUM: Okay, this is Nico Zweibaum again, director of salt systems design talking about the inventory management system. Again, to clarify, this is salt inventory.

That system, IMS, is not credited with performing any safety-related function. What its function is, is to maintain primary coolant level in the reactor vessel during normal operations. It also fills and drains the reactor vessel and the primary heat transport system of salts during startup and shutdown operations. And it maintains primary coolant purity by replacing circulating salt with new salt.

Due to the proximity of the IMS to the reactor vessel, it's designed so that seismic induced failure does not impact the reactor vessel system consistent with PDC 2. Due to its proximity to the reactor vessel, it's designed so that adverse effects of postulated failures do not impact the reactor vessel system, consistent with PDC 4.

That system includes design features to limit the loss of reactor vessel coolant inventory in the event of breaks in the system, consistent with PDC 33. This system may be used to remove and replace a sufficient amount of reactor coolants to restore performance to the FLiBe specific, consistent with PDC 70.

And this is where the interface with the chemistry control system that I mentioned earlier comes into play. And again, statement around minimizing contamination of facility and the environment and facilitate eventual decommissioning.

CHAIR PETTI: So is this a batch system or is it sort of always operating as an extra tank of FLiBe and somehow automatically you would add as much as you need?

MR. ZWEIBAUM: Yes. So the way that the system architecture is setup, and you can see that in the PSAR there is figure, but there will be a pump mechanism to constantly add salt from the IMS tank to the reactor vessel, and an overflow line from the reactor vessel to return exiting FLiBe from the reactor vessel to the IMS.

CHAIR PETTI: So then Ron's earlier question on doing something, not batching chemistry system, you've got a lot of plumbing here, you could do, you could look at that I guess?

MR. ZWEIBAUM: Yes.

CHAIR PETTI: Okay.

MR. ZWEIBAUM: We could.

CHAIR PETTI: Thanks.

MEMBER HALNON: The system talked about the three different tanks. RV coolant level, fill and drain and PHTS fill and drain. And then went on to say that any one tank could do all three functions.

Is it sized for all three functions or is it just that it could be cross-connected, or, I mean, piped into the system as necessary? How does that work?

MR. ZWEIBAUM: So as mentioned in the PSAR, the three tanks that were shown in there are really intended to clarify three different functions. Whether these functions are going to be served by a single, or two or three separate physical tanks, will be clarified with the operating license --

CHAIR PETTI: Okay. Yes. That wasn't clear in my reading. Thank you.

MR. ZWEIBAUM: Okay. The reactor thermal

management system. That system is not credited with performing any safety-related functions. Nearby safety-related SSCs are protected from RTMS failure in the event of an earthquake consistent with PDC 2.

The reactor thermal management system using water cooling to transfer heat from SSCs to the ultimate heat sink during normal operation and maintains the operational temperature limits of concrete structures during normal operations consistent with PDC 44. The system is designed to permit periodic, appropriate inspections and testing to ensure integrity and capability to cool SSCs and to ensure adequate interface with other systems, supporting heat transfer to the ultimate heat sink, consistent with PDC 45 and 46.

So this was on cooling, but on the heating front, that system is also designed to preheat the reactor vessel, and to ensure FLiBe in the vessel is maintained above the minimum operating temperature consistent with PDC 71. And the system is designed to the extent practical to minimize contamination of its sodium environment and facilitate eventual decommissioning.

CHAIR PETTI: So let me just, I want to be clear, because earlier when we read our memo in on

Chapter 6, Member March-Leuba described it as a reactor auxiliary heating system. But it really is the reactor thermal management system that's there to keep the FLiBe above melting?

MR. ZWEIBAUM: Yes, that's --

CHAIR PETTI: Okay.

MR. ZWEIBAUM: Yes.

CHAIR PETTI: So I will change his words to be consistent. Thank you.

MEMBER HALNON: This system, does it directly support safety-related systems? I mean, the concrete and other pumps may be or other equipment?

MR. ZWEIBAUM: Sorry, could you repeat the question please?

MEMBER HALNON: Yes. I'm sorry, I was unclear. Does this system directly support safety-related equipment? Is it a support system?

MR. ZWEIBAUM: Can you clarify what you mean by directly support? Physically or functionally?

MEMBER HALNON: Well, I don't know what the difference between those are. Not physically from a structural standpoint. Certainly does it remove, does it provide essential cooling to any safety-related systems or structures --

MR. ZWEIBAUM: No.

MEMBER HALNON: -- or does the concrete that it's keeping at temperature, that concrete structure, is it a safety-related structure?

MR. ZWEIBAUM: No. The cooling that's provided here is non safety-related.

MEMBER HALNON: Okay. So no safety-related equipment depends on this system to operate?

MR. ZWEIBAUM: Correct.

MEMBER HALNON: Okay. So therefore the -- and I guess the point of my question was, would there be any future tech specs required for a support system? It sounds like the answer to that is no.

MR. ZWEIBAUM: Not expected at the moment.

MEMBER HALNON: Okay. Thanks.

MR. LINGENFELTER: Hi, everyone. I'm Andrew Lingenfelter. Lead engineer of engineer and integration. I'll be walking us through several of these auxiliary system sections.

The first one I'll be talking about here is reactor building heating, ventilation and air condition system. And first and foremost, the reactor building HVAC is not credited to perform and safety-related functions.

Also, the system provides, the function of the system is providing independent environment control to the reactor building. Um, this system is not, or excuse me, is designed to ensure occupational dose does not exceed 10 CFR 20 limits. And consistent with 10 CFR 20.1406, similar to some of the other systems we've talked through, reactor building HVAC is designed to the extent practical to minimize contamination of the facility and the environment, and to facilitate eventual decommissioning.

Some of the -- was somebody providing a comment there? Okay.

Some of the design basis here. The system does not adversely affect safety-related SSCs located nearby in accordance with PDC 2.

In accordance with PDC 60, the system is designed to control the release of radioactive materials and gaseous effluents during normal operation.

And in accordance with PDC 64, the system is designed to provide for monitoring the reactor building effluent discharge bypass for radioactivity that may be released during operation.

MEMBER HALNON: But during postulated events is it isolated and not released into the environment?

MR. LINGENFELTER: The reactor building

HVAC system does not perform a safety-related function in that regard. Whether that's performed for other means, we'll be working on that towards the operating license.

MEMBER HALNON: Okay. I was just kind of reacting to your last bullet there. That may be released during normal operation. Of course the next question is, what about, may be released during a postulated event?

MR. LINGENFELTER: Is your question, will there be releases during postulated events through that system?

MEMBER HALNON: No. Will there be releases through the HVAC system or the HVAC?

MR. DENMAN: So, this is Matthew Denman, the responsible engineer for radiological source term. As part of the Chapter 13 analysis we have to exam a number of postulated events. One of those postulated events is released of radioactive material from subsystems or components at any radioactive material that could be stored within the HVAC system would be considered in that postulated event.

MEMBER HALNON: Okay. So it's, as you get through all your event analysis, this will pan out one way or the other? Okay.

MR. DENMAN: Correct.

CHAIR PETTI: I have a question. Is this also the system -- is there going to be something to monitor beryllium in the facility?

MR. LINGENFELTER: There will be.

CHAIR PETTI: It will be part of this system?

MR. LINGENFELTER: Those are the details that we will be working towards for the OLA.

CHAIR PETTI: Both the monitoring and if one needs to trap beryllium particulates. I'm just worried, you know, if you know what happened at MSRE, you got dendritic beryllium fluoride at cold spots, so valve stems like off the pump, and that, it was dendritic so they went airborne pretty easily. So we'll have to monitor that because, you know, the beryllium limits for workers are really tight.

MR. LINGENFELTER: Got it. Thank you for your comment.

MR. ZWEIBAUM: Okay. Nico Zweibaum, director of salt systems design. I'm going to walk you through our pebble handling and storage system. This system is responsible for handling of fuel of Hermes from initial onsite received through in process circulation until final onsite storage.

Because of the relative complexity of this system, we wanted to break it down into major components here that we'll be walking through. We have a pebble extraction machine. That's a single screw mechanism that removes pebbles from the molten salts in the core.

We have a pebble inspection subsection that performs flaw detection and burn-up measurements of the removed pebbles. A processing subsystem that will sort pebbles into appropriate buffer storage channels based on pebble type. An insertion mechanism that's a separate wheel feeder mechanism that inserts pebbles into the reactor through an in-vessel insertion line.

We have storage canisters. Each canister stores about 2,000 fuel pebbles in a non-critical configuration. We have a storage cooling area that's in-building storage for spent fuel canisters that's capable of passive cooling during loss of power and other postulated events. And we have a new pebble introduction subsection that will store fresh fuel and prepare fuel for circulation with a high temperature bake out prior to fuel introduction into the salts.

Next slide. So this kind of illustrates with the relevant connections. The architecture of the system. I won't go into all details of what's on this slide, but as far as the systems I described earlier, you see your reactor vessel, simplified sketch in the bottom right.

Sitting at the top of that in the center is the pebble extraction machine. Pebbles that get off of that go through an off-head conveyance line to the buffer storage system where there is inspection processing and sorting into different bins.

You can see that if pebbles are sent to storage, we have an active storage canister that's connected to that buffer storage tank. And if pebbles are meant to be reinserted into the reactor, there is an assertion machine that also connects, it's shown here at the top, connects to the buffer storage tank to push pebbles back through a pebble insertion line that is shown on the very right of the diagram here.

I guess of note, and hopefully by now people are familiar with this aspect of our design, but the fuel is buoyant in the salt, which is why we have mechanisms to push the fuel all the way down to the bottom of the reactor. And then the fuel naturally moves up as it gets extracted from the top in the pebble extraction machine. And then on the left of this whole diagram you see our fuel cell that includes the spent fuel storage pool and spent fuel storage air-cooled area.

On the design basis for the system, the storage bay, the pool, and the support restraint structures in the pool are designed as seismic category SDC-3, to ensure the geometry is maintained in the event of an earthquake consistent with PDC 2. By design, this system limits grinding of pebbles and accumulation of graphite dust to minimize the potential of fire and explosion consistent with PDC 3.

The canister design considers environmental conditions, such as pressure accumulation of radionuclides and thermal loads. The interior of the canister accounts for radiolysis products. The system design accounts for complete submergence and internal flooding of the storage canisters in water as part of our criticality analysis, consistent with PDC 4.

There is an anti-siphon feature on the pebble insertion line that limits inventory loss to the primary salt pump elevation. And the pebble extraction machine is above the coolant free surface, consistent with PDC 33 to avoid loss of coolant from the reactor vessel.

The TRISO particle confines radioactive

material rather than the PHSS. And pebble loads do not introduce incremental particle failures thereby ensuring that PHSS does not act to confine or contain radioactivity consistent with PDC 61. And back to the functional containment discussion that we had earlier.

The design prevents criticality by controlling pebble removal rate. The system design precludes moisture intrusion and handling equipment maintains geometry of the interlocks consistent with PDC 62.

And finally, the inspection and sorting function ensures that damaged pebbles are removed from use consistent with PDC 63.

CHAIR PETTI: So I had a question here. I understand graphite pebbles and fuel pebbles, and burn-up is protected using cesium measurement of the pebbles. It's the way the Germans did it. I think the Chinese are doing it. But now that you have natural uranium pebbles in there, is the cesium signal from the fission of the small amount of u-235 going to be enough that you can differentiate between a pure graphite pebble and a uranium pebble? Natural uranium pebble.

MR. WHATCOTT: Hi, this is Gareth Whatcott. I'm the responsible engineer for the pebble handling system. You're correct that the cesium signature early on may be challenging to identify in uranium pebbles. The method to differentiate between the natural uranium and fuel pebbles will be detailed in the OLA.

MEMBER HALNON: It wasn't the natural uranium.

CHAIR PETTI: I suppose you could co-mingle them in your storage, but --

MR. WHATCOTT: Yes, okay. Sorry. Sorry for mis-replying. I think the answer still holds that the differentiation between those two will be something that's detailed in the OLA.

CHAIR PETTI: But right now, at least for the pictures, right, I mean, okay, if it's a regular fuel pebble you've got storage for that, but you're eventually going to have to get rid of the natural uranium pebbles. So is there a separate storage system? How is that all envisioned?

MR. WHATCOTT: No, the storage system will be with the fuel and the natural uranium pebbles will be stored together in --

CHAIR PETTI: Okay.

MR. WHATCOTT: -- special storage.

CHAIR PETTI: Okay. Thanks.

DR. SCHULTZ: This is Steve Schultz. I have a question here on the criticality evaluations. In the PSAR there is a couple of places where you do indicate, as shown on the slide here, that the criticality analysis is performed, assuming both that the storage bay and canister interiors are flooded and the evaluation is done that way for conservatism. But later on when you talk about satisfying PDC 62, you indicated that the criticality analysis of preliminary one has been performed with the canister flooded, but the interior of the canister is not flooded. Is there a reason for that or are you looking for, I'm looking for consistency between the assumptions there and I was surprised to see the inconsistency.

MR. ZWEIBAUM: Yes, this was an error that we caught. This is indeed an inconsistency. The analysis has been, and will be performed, assuming that everything is flooded. So this is a typo in the version --

DR. SCHULTZ: Good.

MR. ZWEIBAUM: -- of the PSAR that was shared, and will be corrected.

DR. SCHULTZ: Quite satisfactory. Thank you.

MR. LINGENFELTER: Okay, this is Andrew

Lingenfelter again. Lead engineer of engineer integration. I'll be talking about the fire protection systems and programs.

So, first off, the fire protection system is not credited with performing any safety-related functions. In accordance with PDC 3, it's designed to protect and extinguish fires so that a continuing fire will not prevent safe shutdown.

Also in accordance with PDC 3, non-combustible and fire-resistant materials are usual never practical. Particularly in locations with SSCs that are safety-related or required for safe shutdown.

Some of the codes that the fire protection system will conform to are the local building and fire codes, ANSI-ANS 15.7, fire protection program for research reactors, NFPA 801, and life safety code NFPA 101. Also in accordance with PDC 2, the system is designed so that seismic induced study does not impact nearby safety-related SSCs. Okay.

DR. CILLIERS: Thank you. This is Anthony Cilliers speaking. I'm the director of instrumentation and controls. And I'll be talking about communication.

I'd just like to note here that this is communication technologies that is provided between humans. This is independent of our communication systems, the data communication throughout the control systems, as well as the calculation of data. So this is purely communications between people onsite and offsite.

The communication system is upgraded for performing any safety-related functions. And we've identified a coupled of technologies to allow us to have the appropriate type of communication for whatever case is required. And these technologies will do things like plant radio, public address and general alarm system, communication capability in the event of loss of normal power, so we can still communicate in the event of loss of power, distributed antenna communication and security communications.

In addition to this we will be using diverse commercial communication systems for on and offsite communication. And that allows just additional layer of diversity for wireless communication we needed between staff onsite, as well as provided information offsite.

It's used for normal and emergency communications to communicate between the key areas of the facility. And will also be provided phone lines for offsite communication in case of an emergency.

This is fairly conventional communication systems as per normal in nuclear power plants. Thank you.

MEMBER BROWN: When you say conventional, most of this electronic and wireless that you've listed in here. A phone line varies in what it can be. It can be the loss of, like, fiber optic phone lines will die when you lose power. Are you going to have any, what I would call conventional land lines available, the old style, such that if you lose power you can still communicate, or are you going to --

DR. CILLIERS: Yes. We will have the conventional phone lines installed. Just as it is the easiest system as a backup system. But we'll be utilizing most of technologically advanced wireless systems where appropriate because it allows us additional level of freedom for communications.

MEMBER BROWN: Yes, I don't have a problem --

(Simultaneous speaking.)

MEMBER BROWN: Yes, I don't have a problem with that. Okay. I just, hardwired, you know, copper land lines, a couple of them would be very, very good to have just in case. And they're easy to put in.

If you depend on a fiber optic-type phone line, well, then you're dependent upon loss of power also and have to have backup power to make sure you can keep it. Most of those won't last for most of 24 hours, as I can attest in my house.

DR. CILLIERS: Yes. Well --

MEMBER BROWN: It's just a thought. I mean, I have no problem with the other stuff you're doing, it's just that whatever phone lines you have, if you ought to have some of the old-style copper phone lines, which go out, which are self-powered.

DR. CILLIERS: Yes.

MEMBER BROWN: And they'll always be there for you. I wouldn't use them for the general stuff around, you're obviously going to do it the other way. Which is fine.

DR. CILLIERS: Yes, thank you. The last bullet, the intent of that last bullet really is to cover all of the valves and normal copper phone lines.

MEMBER BROWN: Okay, thank you.

MS. ELLENSON: Hi, this is Margaret Ellenson. I am the safeguards and security manager here at Kairos Power. I'm going to be covering Section 9.6. There are various materials that we're aware of that will be on the site at Hermes. We'll have byproduct material, source term and special nuclear material.

So this section covers our expectations related to how we are going to license the use of those materials at the site. In particular, we're going to be applying for the traditional 30, 40, 70 license. That is 10 CFR Part 30, 10 CFR Part 40, and 10 CFR Part 70 for byproduct source of special nuclear material respectively.

We are actively engaged with the staff about the sequencing of when they need the information to support those licenses. And our expectation is that the licenses will be provided as part of the operating license application. Any questions about 9.6?

MEMBER HALNON: Yes. That last bullet request, will be submitted at a future date. Is that what you just mentioned is, these will specifically be part of the operating application?

MS. ELLENSON: Yes. And I think the CPA specifically says, the licenses will be submitted at a future date. There is some complexities about when information will be available, depending on when various program elements are available, right? Those program elements that are needed to support the Part 30, 40, 70 applications. So we are actively engaged on like how to get the information to the staff at the right time to support the reviews.

MEMBER HALNON: Okay. Yes, there is a lot of admin in here.

MS. ELLENSON: Yes.

MEMBER HALNON: Yesterday we had a discussion about the short operating license, decommissioning windows and whatnot. And in Chapter, I think 17, you mentioned there is no intent to get a material or possession only license later on.

I know that that this, it just may not be in this section, but it just talks about, it's a material license, basically, or a special nuclear material license. Just kind of keep that in mind. I think that's important to assess whether or not you will need a possession only license after the four-year period.

MS. ELLENSON: That's a good point. Thank you.

MR. LINGENFELTER: Okay. This is Andrew Lingenfelter again and I'll be talking about the auxiliary systems in 9.7 here and 9.8.

So the first one is a plant water systems. The water systems are not credited with performing any safety-related function. The first of those systems is a service water system. And this is the system that facilitates moving the main supply of water that we received and providing that water to the treated water system. Along with other areas.

The treated water system provides chemistry control of the service water and provides water to the component cooling water system, chilled water and decay heat removal systems. In accordance with PDC 2, the system is designed to protect against design basis earthquakes, nearby safety-relates SSCs.

Also in accordance with PDC 4, nearby safety-related SSCs are protected from the effects of discharging fluid and missiles, and precluded from pipe whip hazards by design.

The component cooling water system, this provides water cooling for the reactor building HVAC systems, the equipment instructional cooling system, plant chilled cooling system, and the inert gas system coolers and compressors. And also in accordance with PDC 2, this one will protect against design basis earthquakes for nearby particular SSCs. And then also it will follow the guidelines of PDC 4, or nearby safety-related SSCs that are protected from effects of discharging fluid and missiles, and precluded from pipe whip hazards by design.

Also, the system is designed with a

capability to isolate leaks, permit appropriate periodic inspection and testing, to ensure the integrity and capability of the system to cool SSCs and to adequately transfer heat to the ultimate heat sink, in accordance with PDCs 44, 45 and 46.

The chilled water system provides cooling water for nonessential heat loads. And consistent with 10 CFR 20.1406, the plant water systems that directly interface with systems that contain radioactive material are designed, to the extent practicable, to minimize contamination of the facility and the environment and to facilitate eventual decommissioning.

The next one I'll be talking about here is the remote maintenance and inspection system. This one is not crediting with performing any safety-related functions. And the system can remotely handle components in the reactor systems PHTS and PHSS. The system supports the following maintenance activities. And I'll actually let you read those. I don't necessary have to go through each of those.

And consistent with 10 CFR 20.1406, as we've been discussing in a lot of these slides, that the system is designed, to the extent practical, to minimize contamination of the facility and the environment and facilitate potential decommissioning. The capabilities of this system will limit personnel occupational exposures to below 10 CFR 20 limits.

Okay. The next one, spent fuel cooling system. This one is not accredited with performing any safety-related functions. What the functions it does provide is the forced air cooling for spent fuel storage canisters in the storage bay of the PHSS, and recirculates water in the spent fuel pool. And this consists of fans and pipings that remove heat during normal operation. And maintains desired operation temperatures in the storage bay.

And if power, normal power is not available, the system is capable of passively cooling the spent fuel storage canisters. And some of the design bases here, in accordance with PDC 2, the system is designed to ensure nearby safety-related SSCs that are protected from seismic induced failure. And in accordance with PDC 4, nearby safety-related SSCs are protected from dynamic effects, such as missiles by design.

And a similar statement is, previous to 10, it's consistent with 10 CFR 20.1406, system to the extent practical, will minimize contamination of the facility and the environment and facilitate eventual decommissioning.

Okay. Next one, compressed air system. This system is not accredited with performing any safety-related functions. What the system does, is providing and distributing compressed air for maintenance and use for in-valve operation. And the system is designed so that a failure of that system does not interfere or preclude the ability of a safety-related system to perform its safety function. And this system does not directly interface with systems that contain or have potential to contain radioactive materials.

Okay. Next one is cranes and rigging. So again, this is not accredited with performing any safety-related functions. What this system does is it will be lifting and moving equipment within the reactor building, facilitate receiving and shipping and supporting maintenance activities.

And in accordance with PDC 2, the system will ensure nearby SSCs, are safety-related SSCs that are protected from seismic induced failure. And in accordance with PDC 4, nearby safety-related SSCs are protected from dynamic effects by design, such as administrative controls and interlocks. And this will implement codes and standards from ASME B30.2.

Okay. Next one is auxiliary site services. This is not accredited with performing any safety-related functions. The following services are provided, you can read there by the system, but I won't go through each of those.

In accordance with PDC 2, the system is designed to ensure nearby safety-related SSCs are protected from seismic induced failure. And the capabilities of the system will limit personnel occupational exposures to below 10 CFR 20 limits.

In services that involve handling the radioactive material may include remote manipulation capabilities, as appropriate, to facilitate limiting personnel occupational exposures to believe 10 CFR 21.

I think that's the end of our slides.

CHAIR PETTI: Members, any questions? Okay, then let's turn to the Staff.

MEMBER DIMITRIJEVIC: That, Dave, can you hear me?

CHAIR PETTI: Oh yes. Yes. You had a question?

MEMBER DIMITRIJEVIC: I'm sorry, I have a really bad technical setup here so I couldn't unmute myself for a second.

I had a general question about this

auxiliary systems and, and Greg asked in one moment, do they support any safety questions. So this is my general question. When we say that those systems are not accredited in performing any safety function, which is always the full sentence in presentation, does that mean they are not supporting any safety functions, too, and that they're failure would not impact any safety function?

So does this first sentence implies, where they say not accredited to performing safety function, does it also imply they're not supporting any safety function and their failure would not impact any safety function? That's my general question.

(Simultaneous speaking.)

MEMBER DIMITRIJEVIC: But second question is, was there any something like ASME, you know, failure mode and effect analysis performed on this systems where you analyzed for type of the failure modes they can be and how they affect the plants? Was that a part of, like, for example, analyzing postulated accident?

That's my second question. And my -- and I noticed, actually, that often we talk about dynamic -- for subsystems, we talk about those dynamic effects on the safety SSCs. But that's not -- like, for example, I know from the fire analysis in the Section 3 that we actually left for the operating license to see how can that water source be ethylated (phonetic) so it will not effecte that.

Is there any more detailed documentation on the type of this secondary or the economic effects on these systems? Actually, everything is connected that was -- how and when those systems analyze for their failure and impact.

MR. LINGENFELTER: Okay, well, I can -- this is Andrew Lingenfelter. Thanks for those questions and comments there. I'll go ahead and field the first one, and then I'll have our director of reliability handle the second one.

So the first one, I believe your question was, by saying that it does not perform any safety-related function, does that also mean it does not support any safety-related functions, if I'm not mistaken. And the answer to that question is, yes.

MEMBER DIMITRIJEVIC: Well, how then this -- because we have argued yesterday the FLiBe and foil are the safety components. And obviously some of those systems support that, you know, like temperature control, the FLiBe and things like. Why would you say they don't support the safety functions then?

MR. HAGAMAN: So this is Jordan Hagamanm and I'm the director of reliability engineering. When we talk about whether a function is safety-related we are specifically talking about the performance of the SSC during postulated accident conditions.

So the cooling functions that happen during steady state, there are cooling functions for safety-related systems, but the way we provide assurance of the operability of those SSCs is through a limiting condition of operation that will be in the tech spec. Where temperature is important for the operability of a system, we'll be monitoring the temperature as part of our limiting condition of operation.

So the actual cooling function is provided in steady state, it's not a safety-related function. We rely on the tech spec to ensure that the SSC is operable. And then the cooling function is not required during the postulated event mission, mission time.

MEMBER DIMITRIJEVIC: Okay. But that is -- also part of my suspicion was, does that mean the system failure would not interfere?

See, that is bad thing because if you, the system failure will not interfere with performing safety function.

MR. HAGAMAN: Yes. So that is something that we also check against to ensure that the system cannot fail in a way that impedes a safety function.

MR. GARDNER: Okay, this is Darrell Gardner. This is why we have the PDC 2 and the PDC 4 specifically mentioned for most of these systems. So for example, if you have one of these non-safety systems routed closely or in proximity to something that was safety-related, then we're going to restrain it appropriately. Or put up a barrier, or ensure the distance is sufficiently far enough away that it can't have an adverse interaction.

But until we do specific layout and routing of those systems, we can't tell you which ones, or which portions of those systems might possibly have a failure mechanism such as, again, failure due to seismic or failure due to pipe breakage for PDC 4.

MEMBER DIMITRIJEVIC: Okay. And I have a really, I mean, we're going to think a little more about all of this in Section 13. And how about failure mode and effects analyses, do perform that for any of those systems?

MR. LINGENFELTER: We regularly perform

failure modes and effects analysis in conjunction with the level of design detail available at any stage. And we regularly update those failure modes and effects analyses as the design progresses.

So we can expect that as we get final design to support the operating license we'll have a final more static version of the failure modes and effects analysis that confirm the safety functions, and in particular, that the non-safety systems don't, cannot impede any safety functions.

MEMBER DIMITRIJEVIC: And was that a part of the review?

MR. LINGENFELTER: It was available for review. I'd have to go into my notes to see how much we actually looked at it with the staff.

MEMBER DIMITRIJEVIC: Okay, thanks.

MEMBER REMPE: Maybe the Staff should answer that question.

MR. LINGENFELTER: Yes.

MEMBER REMPE: Did you look at some of their FMEAs as part of your review? And could you talk a little bit about the details since they said it was commensurate with the detail of the design?

MR. BEASLEY: So this is Ben Beasley with the Staff. I reviewed just the reactor build in the

HVAC, I did not look at a modes and effects analysis. I don't know if there were four other reviewers in Chapter 9.

MEMBER REMPE: So you don't remember seeing --

MR. BEASLEY: I don't remember if any of them pursued that or not. I don't remember specifically.

MEMBER REMPE: Do you remember even seeing if they were submitted?

MR. BEASLEY: Well they wouldn't have been submitted --

MEMBER REMPE: Oh.

MR. BEASLEY: -- we would have done it through audit.

MEMBER REMPE: Audit, yes.

MR. BEASLEY: Yes.

MEMBER REMPE: That they were available to you, you used the audit.

MR. BEASLEY: Yes. If we had wanted to see it, it would have been put on --

MEMBER REMPE: You didn't recall seeing any?

MR. BEASLEY: I do not recall seeing any. No, ma'am.

MR. CHERESKIN: So this is Alex Chereskin from the NRC Staff. I, again, don't recall about the, you know, specifically reviewing failure modes and effects analysis, but just as an example, with the inert gas system we did ask about the consequence of, you know, what happens if air gets in here. Because obviously it's connected to the handling system, you can have oxidation. So we did have those discussions.

And the resolution ends up that it's bounded by Chapter 13 analysis. So, you know, at least when I looked at it up there, just as an example, it wasn't explicitly an FMEA, but we were considering what happened is if you have a failure and what's the data, the effects of that.

MEMBER REMPE: So they thought about it but didn't see documented analysis. Okay, thank you.

CHAIR PETTI: So, it just, it seems to me that, I think of the system in other technologies the supporting system is playing an important role in ensuring the safety functions. But here at least the critical safety function of functional containment is inherent in TRISO, assuming you don't get above really high temperatures, which they have huge margins, and FLiBe, it's inherent in the nature of FLiBe that the fission products have attained whether it's spilled, whether it be in a tank. And so, I think because of that there is a lot less coupling of the support systems from the systems that do accomplish the safety functions.

It's, again, I think it's another one of these things where the functional containment evolves in the design in a different way of other technology. Any other questions, members? If not we'll move, go on to the staff.

MR. BEASLEY: Thank you. Again, this is Ben Beasley, I'm with the NRC Staff. And I do want to introduce the other presenters for Chapter 9. Alex Chereskin will talk about a couple of the auxiliary systems that he reviewed. And Jason Schaperow is joining us virtually. He'll talk about the systems that he reviewed.

And I would like to do a quick mic check for Jason. Are you present, and can you check your mic?

MR. SCHAPEROW: Yes, I am.

MR. BEASLEY: Very good. Thank you. So, it's already gone to the second slide. This lists the content of the safety evaluation and the PSAR. Nothing in the auxiliary systems of Chapter 9 is safety-related, as you have already noted.

We're going to focus on the systems that are novel and that have more relevance, yes, relevance to safe operation. Those systems are primarily in 9.1 and 9.3. If you have any questions on other systems, HVAC, fire protection, communication, use of nuclear material or other miscellaneous systems, those reviewers are here and would be able to answer your, or I'll try to answer your questions.

So the next slide.

MEMBER HALNON: Ben, you mentioned that --

MR. BEASLEY: Yes.

MEMBER HALNON: -- nothing is safety-related. I understand there is no safety-related functions, but the pump handling the storage system, I mean, that's got FLiBe in it and pebbles. Is that not designed as a safety-related, I mean, piping and pressure boundaries and stuff like that?

MR. BEASLEY: So Jason is going to present on the PHSS in a few minutes.

MEMBER HALNON: All right. So I'll just queue up that question.

MR. BEASLEY: Yes.

MEMBER HALNON: Thank you.

MR. BEASLEY: But Jason will be ready to answer.

MR. SCHAPEROW: Yes. Yes. I'll pull up my notes because I actually talked to the Kairos about that during an audit meeting.

MEMBER HALNON: I'll wait. Thanks, Jason.

MR. BEASLEY: So, Kairos did not request approval of any specific auxiliary system designs in their construction permit application. The application contained a description of the preliminary design and identified the design bases.

In general, the evaluation criteria required, NUREG-1537 required that auxiliary systems should not result in reactor accidents or uncontrolled release of radioactivity and that no function or malfunction of the auxiliary systems should interfere with or prevent safe shutdown of the reactor. The Staff review determined that the preliminary designs of the auxiliary systems are consistent with the objectives of the identified PDCs and NUREG-1537, thus the information on auxiliary systems meets the requirements of 10 CFR 50.34 for a construction permit. Further information needed to complete the safety analysis can reasonably be left with the operating license application.

I will turn it over now to Alex, who will start talking about the chemistry control system.

MR. CHERESKIN: Sure. This is Alex Chereskin, the NRC Staff. I'll be covering sections 9.1.1 for the chemistry control system and 9.1.2 for the inert gas system.

So starting with the chemistry control system, just a brief overview. It's used during normal operations to monitor coolant chemistry, ensure the FLiBe meets appropriate specifications. It would also be able to remove or replace the coolant to restore performance specifications via the connections to the inventory management system that Kairos described earlier in their presentation. And has been just noted here, this is a non-safety-related system.

Next slide, please. So this slide contains the Staff evaluation of the chemistry control system. And so, I'll start with the PDC 2, requiring protection against natural phenomena. As you heard yesterday, that seismic methodologies were talked about in section, Chapter 3 of the Staff SER. And, you know, this is also something that we would look at the operating license stage, as Kairos mentioned, you know, once the layouts are finalized to make sure that failure cannot impact the safety-related SSC.

And I think it's pretty similar for PDC 4. It's the similar thought protecting against environmental and dynamic effects. Once we can see that final layout, I think that's something that could be verified.

In addition, PDC 70 requires that a chemistry control system requires a system to monitor and correct the reactor coolant chemistry. And PDC 70 basis that on factors such chemical attack, fouling and plugging, radionuclide concentration and the potential for air and moisture ingress. So there is reasonable assurance that the chemistry control system will be able to measure purity and correct the chemistry via the IMA if needed.

As Kairos talked about in their presentation, if the samples are found to be out of specification, what would be replaced in order to restore performance of the specifications. And at the operating license application stage, the Staff would review the, you know, where the samples are taken in the CCS to ensure that you have a well-mixed representative sample of the coolant.

And we would also review the other methods used to analyze the coolant. I think that's something that some of the members were talking a little bit about before. As well as the frequencies. And, you know, if there are any corrective actions required if you go add a specification.

And so, you know, the CCS is also consistent with some of the proposed limits that were found in the proposed limits in Chapter 14 of the PSAR talking about things like maintaining the correct lithium fluoride to beryllium fluoride ratio, as well as maintaining appropriate circulating activity limits.

And one other criteria that I wanted to mention here, there is a NUREG-1537 guidance about ensuring that whole cleanup systems don't cause an excessive loss of coolant in the other CCSs downstream of the anti-siphon features in the vessel. And so, it would seem unlikely that the CCS could cause that excessive loss of coolant.

Next slide, please. So now we'll cover the Staff's review of the inert gas system. And per the PSAR Section 9.1.2, the IGS has several design functions. That includes maintaining an inert environment, providing purge flow and removing impurities, as well as transporting tritium downstream to the tritium management system. And also providing the reactor coolant motive force. It also needs to be able to assess the purity and process the gases, as well as ensuring control and protection of leaks from the IGS.

The Staff has reasonable assurance that criteria in NUREG-1537 can be met as the IGS will use an inert argon gas that's supplied to the components with individually regulated temperatures, pressures and flows. It will contain storage tanks, as well as equipment to be able to measure and remove oxygen and moisture. And it will also, it also can be monitored for leaks. And it will contain radiation monitors, as shown on the P&ID in the PSAR.

Next slide, please. And so here, again, we have PDCs 2 and 4. And so, I think similar to some of the previous discussions we've had, once the, you know, the final plant layouts are determined, we would be able to verify that failures of these systems, you know, from like seismic or dynamic effects wouldn't impact safety-related systems.

In addition, PDC 64 would require the monitoring of radioactive releases. And as I noted on the last slide, the IGS does have radiation monitors and can be inspected to detect leaks. And so that would help to meet PDC 64. As well as the proposed technical specification on circulating activity.

And additionally, PSAR Section 9.1.2 talks about assessing argon volume purity, which is consistent with the NUREG-1537 guidance that the gas purity should be assessed.

And the other thing, it's not on the slide, but because we're talking about the potentials for how these support systems may impact other things and whether we've looked at, you know, some of the things that we discussed during the audit, in addition to the one I mentioned before was that, you know, the IGS was also used to ensure that FLiBe doesn't freeze in certain areas. And, you know, we talked about that in being able to provide kind of like a sweep gas. When you move any like FLiBe aerosols along to help try and mitigate that.

And we talked about the air ingress potential just a little bit before. And then one of the other things that was discussed during our audit discussions were, you know, whether or not like a high initial pressure in the IGS could have impacts on the natural circulation flow, but given that the FLiBe level would be, the natural circulation would take place below the FLiBe free surface, you know, it would seem unlikely that the IGS would be able to impact that.

So, you know, just based on the previous discussion I wanted to give at least a couple of examples of areas where we did consider that, even though these systems were classified as non-safety-related. And I think that's the last slide I have on the IGS.

MEMBER HALNON: Yes. And, Alex, thank you for going through some of the behind the scene questions. That helps give us confidence in that your review is pretty comprehensive, so I appreciate that.

MR. CHERESKIN: And so, if there are no further questions I'll turn it over to Jason because I believe he has the next session. Section.

MR. SCHAPEROW: Yes. I'm Jason Schaperow with the NRC Staff. Today I will be presenting the Staff's review of some of the reactor coolant auxiliary systems. As well as the Staff's review of the fuel handling and storage system.

So as noted on this slide, the three-reactor coolant auxiliary systems that I reviewed were the tritium mitigation system, the inventory management system and the reactor thermal management system.

So these are kind of unique systems. And as such, when I looked at the review guidance for test reactors there is not a specific section for tritium mitigation. That's not one of the sections.

What they do have is, they have a section, Section 9.7, called other auxiliary systems. And this gives review criteria and guidance for these types of systems, which are other auxiliary systems, which are not safety-related.

I tried to summarize here in just three short bullets. These systems should be such that they do not result in reactor accidents. They do not prevent safe shutdown. And they do not result in unacceptable releases or exposures.

Next slide, please. So regarding our review, what we did was we asked lots of questions at audits to enhance our understanding of what the texts that's in the PSAR. We checked the PDCs that are listed in the PSAR, which actually Kairos showed in their slides. We checked those to make sure those are appropriate to meet the review criteria.

And finally, for these facility unique systems we had checked to make sure the PDCs listed are appropriate for the specific system. There is a specific PDC for the thermal management system. For example, it's called PDC 71, reactor coolant heating systems.

Next slide. Regarding fuel handling and storage, there are review criteria given for this.

Actually, this section number is not listed here, sorry. The title of NUREG-1537 is not handling and storage it's spent fuel. But there should be a section listed.

But anyway. So Kairos calls their system, they gave it a special name, it's called the pebble handling and storage system. This is their handling and storage of spent fuel. So there are a number of review criteria for NRC Staff when reviewing test reactors.

Six is listed here. The first one is that all spent, all special nuclear material must be accounted for. Fuel meets procurement specs. Criticality is prevented. Fuel handling tools and procedures need to be appropriate designed. Methods to assess fuel radioactivity and potential exposure rates need to be adequate to avoid personnel exposure. And finally, they have to deal with their occupational exposure limits in the ALARA.

Next slide, please. Again, with regard to the NRC Staff review, we had meetings with them. We had audit meetings with them to enhance our understandings. And again, we checked the PDCs that Kairos listed in the PSAR to make sure they were the appropriate ones.

With regard to the Members question about what part of the system is considered safety-related, so we did, actually, another reviewer identified that. Ed Helvenston. He noted, hey, Jason, you know, it says here that there is some stuff related, you know, are you sure you understand exactly what is. We did discuss this with Kairos, and they clarified what is safety-related in the system.

And what is, is the concrete structures. I guess I should probably read the text here just to make sure I don't goof this up because it's been a long time since I looked at this. Yes, Kairos is responsible with the concrete structures associated with the storage bay for the canisters, the spent fuel pool and the support restraints in the pool. Those are safety-related structures.

And the reason they are is to ensure that the geometry is maintained to preclude inadvertent criticality during an earthquake. They also mentioned that the pebble extraction machine trip is a safety-related function.

The other things that I think are addressed through Chapter 13 analysis, I recall there being an analysis in Chapter 13 of a break in the pebble handling system where pebbles would spill out onto the floor of the room there where the system was located. And so I think that's how the rest of this is treated. I hope that answers the question.

MEMBER HALNON: Yes, I think it does. I have to think about it, but I don't see any additional follow-up I need.

MR. SCHAPEROW: Over to you, Ben.

MEMBER DIMITRIJEVIC: Sorry, I am again late. Can you go back, I think it was Slide 67. This is Vesna Dimitrijevic.

That when you define that, when was that, when they defined that based on the NUREG, what the three question, not after the reactor shutdown, what, let me just think.

MR. SCHAPEROW: Oh, that's Slide 68 maybe. 68.

MEMBER DIMITRIJEVIC: Yes, somewhere around that. I don't remember exactly what slide.

MR. SCHAPEROW: Yes, there is it.

MEMBER DIMITRIJEVIC: Okay. So let me just see it. So not the stopping the reactor accident, right? That's the question?

MR. SCHAPEROW: Correct. That's what's --

MEMBER DIMITRIJEVIC: We know here that also there is no accidents, right? They have a postulated events, right?

MR. SCHAPEROW: Well, the point, I guess the point I was trying to distill down here was that failure of these systems should not result in a reactor accident. Like if the tritium mitigation system stops functioning or gets a hole in it or something, it won't result in core damage.

MEMBER DIMITRIJEVIC: Oh, okay. Well the reactor accident means will not result in the core damage or will not result in the initiating event?

MR. SCHAPEROW: Hm.

MEMBER DIMITRIJEVIC: Because, seem what happening here since they merge, after the accidents in the coolants and accidents, everything in postulated event, is there a safety question that's in failure of this system would not result in a postulated event. So that's what -- I'd just like a sort of, you know, the trying to understand how this all comes to the postulated events. That's why I'm asking this.

MR. SCHAPEROW: Yes, I don't think NUREG-1537 uses the term postulated events, although I guess it's --

MEMBER DIMITRIJEVIC: I know. But in the Section 3 you have determined that we're going to use this in this application, so --

CHAIR PETTI: Yes, I think the question, Jason is, NUREG-1537 uses a certain set of terms and terminology, but things have been redefined in Kairos, so how did you map what --

MR. SCHAPEROW: I --

CHAIR PETTI: -- 1537 says in terms of the language that Kairos uses?

MR. SCHAPEROW: Yes, I would say they're synonymous in reactor accident or postulated events. I mean, I --

MEMBER DIMITRIJEVIC: Well in that case, if and accident is in the coolant some of those systems would result in shutdown of the reactor, right? And that's also postulated event. I'm just being, you know, interested.

MR. SCHAPEROW: Yes. I don't see that any of these systems, at least the way they're described, would result in reactor shutdown. Again, if the tritium mitigation system stopped working, you might see a little buildup of tritium somewhere in the plant, but it's not going to initiate a reactor --

MEMBER DIMITRIJEVIC: But management, maybe the fuel handling. You know, that's why I think the failure mode doesn't affect analysis --

MEMBER KIRCHNER: Well, I would point out --

MEMBER DIMITRIJEVIC: -- bring this reactor accident information offline. I'm okay.

CHAIR PETTI: Go ahead, Walt. You had a question?

MEMBER KIRCHNER: No. I was just going to say, in tech specs of the applicant in Chapter 14, it identifies classes of limiting conditions of operations that are mainly maintained by the systems at particularly 9.2. And if those, if the detected conditions are outside that envelope than that would have a, certainly have an impact on reactor operations. For example, purity of the purity spec that they need to maintain for the FLiBe.

MR. HELVENSTON: This is Ed Helvenston from the Staff. I'll just, maybe I'll clarify one thing a little bit in terms of the reactor accidents and postulated events terminology.

I think the term postulated events is probably a little broader than accidents. I think accidents may mean something that's actually occurring that has some potential failure or consequence for the reactor.

We do look at a wide range of postulated events in Chapter 13. For example, I think a malfunction of equipment is a possible postulated event, but, you know, we have an analysis in Chapter 13 that demonstrates why that won't result in any unacceptable consequences for the reactor.

So I think there is a little bit of a nuance between those terms. I just, hopefully that clarification is helpful.

MR. SCHMIDT: Yes, this is Jeff Schmidt from the Staff also. So, like things like, let's just say a cooling water system and an interaction with FLiBe is prevented by design. So like you could have a leak of water somewhere, right, and that could be "an initiating event," but it's interaction, it's negative interaction would be precluded by design. So I guess you could call that an initiating event, but it doesn't lead to anything.

I would also agree with Dr. Kirchner that most of these systems set the initial conditions for the postulated events, right? These mostly go into tech specs that set the allowed range of initial conditions, and then the accident, or event, I'm sorry, the event progresses, right? And that sets the initial condition for these systems. It's not used to mitigate the event but sets the initial conditions.

MEMBER DIMITRIJEVIC: But, okay. But my, I'm just, you know, in terms of the NUREG-1537 you're not -- you're integrating this first thing, is not result in postulated accident. Postulate event, sorry. I mean, you know, is that, because here we have this little issue with the implementation of the postulated events, so.

MR. SCHAPEROW: Yes. That's how we're interpreting that, not to result in postulated events.

MEMBER DIMITRIJEVIC: It was obviously just agreeing that that would result in several postulated events, right? And the shutdown is a postulated event.

MR. BEASLEY: So this is Ben Beasley with the Staff. Jason, I'm going to amend your answer there a little bit. I think that we were using this term because that's the terminology used in the criteria in NUREG-1537. So that's why it's showing up in our slides because that's the criteria we were measuring against.

And as Ed pointed out, there is a nuance between what you would consider a postulated event and what we would consider an accident. And so, you know, so weren't examining these systems to assure that they wouldn't create a postulated event, but that they wouldn't create an accident that could have radiological consequences.

And again, this is a very preliminary design and so we couldn't dig into detail. That is what we will do on the next round with the operating license application.

MEMBER DIMITRIJEVIC: Well, maybe even need to, you need to add additional definition here in using this, you know, so there no confusion here.

MR. BEASLEY: Thank you for the comment. And we'll certainly consider that as we move forward with the OL application review. Ed, could you go on to Slide 72.

So unless, you know, I don't want to cut off questions, but we just have one more slide and would be glad to take any of your questions. So the Staff found that the auxiliary system designs, and this is kind of why I wanted to get to this point, we were reviewing against the criteria in 1537, and so the Staff found that auxiliary system design should not result in reactor accidents, or uncontrolled release of radioactivity, and that no function or malfunction of the auxiliary systems should interfere with, or prevent, safe shutdown of the reactor.

And mere design of the auxiliary systems is sufficient and meets the applicable regulatory requirements and guidance for the issuance of a construction permit in accordance with 50.35. Further technical or design information required to complete the safety analysis can be left for later consideration in the FSAR. So that concludes our presentation.

CHAIR PETTI: Members, any additional questions? Okay, then let's move to the memo. Greg.

MEMBER HALNON: Okay. Since there was such a broad topic of space here, the background just basically summarizes the heat system in a principal function of the system. And that was to keep some context of where we were.

If you go to the SCR summary. So, it just reiterates that we agree with the conclusion that there are sufficient details there to provide competence in the evaluation, and we really do appreciate the extra context that you guys have put behind some of the questions you asked.

And we acknowledge the fact that there is a lot to do in the operating license. So, you know, we talked a little bit about how much there is and schedules and that stuff. It's going to be a tremendous, a tremendous effort to schedule that out and get it done as well.

On the discussion, I call out the fact that the Part 20 stuff still has to get done. The items to highlight I'll just kind of step through. I'm not sure, I think you got the same numbers I have. Yes.

The numbers, or the systems described, they have no safety-related functions, as we talked about. But we did find out that there is some safety-related aspects in the construction and equipment. But there is no mitigation functions. And I did come off the pebble handling and storage system, as we mentioned, that there are some parts in there that aren't constructive to safety-related functions.

The PHAs pebble handling storage system, the assumptions I called out, found out that that was an error in the assumptions and criticality analysis. I'll go back, Dave, and revise the memo to make that clear that it was an error and that it will be fixed. But probably a lot of the verbiage will stay there, just so it makes things in context.

I call out the fact that pretty well they have described the fact that non-safety systems at interface are in a general proximity will be designed accordingly with the proper seismic and other designs.

Again, Part 20 requirements needed to be looked at. This anti-siphoning function of the PHTS was, in my mind, is really important for inventory. And when I tried to go through the string of go, talk about it here, go to Chapter 4, then go to Chapter 5, each time it was handed off I expected to see a little bit more detail on what that looked like. And I realized that anti-siphoning is fairly intuitive but I didn't get much details.

So I'll be looking forward to getting that detail a lot to see how that configuration is and how it interfaces. Because that truly is an interface with a safety-related system and the inventory management system. So it's sort of indirect, but it certainly isn't a factor.

CHAIR PETTI: You know, I mentioned it is one of the "novel features" of Chapter 1, so to see it again in another memo is perfect.

MEMBER HALNON: Okay. I did mention that there were, I'm on Number 7 if you're not there yet, there is three tanks. Or at least the picture showed three tanks I believe on the IMS. Inventory management system.

I understand from, and I'll revise this one as well with the comments that we had here that whether it be three tanks or one tank that can do any of the functions, make sure that's clear in here too because I thought that was a clarifying, a good clarifying comment by Kairos.

The low pressure water systems that received water to the plant for cooling, maintenance, potable water, some of them will interface with radioactive systems potentially. The water systems do have that potential so the interconnected system leakage would be a problem. We've seen that in the existing nuclear plants for non-safety, non-contaminated systems that contaminated quite easily with one failure, so we had to make sure that those are recognized.

And then I just mentioned, that there is, in my mind, as the design hold out there could be potentially some technical specification parameters that are necessary to maintain, to put into tech specs for some of these support functions. Maybe not, but again, we don't have the details so let's keep that in mind. The only recommendation was to correct that error in the criticality in the auxiliary, criticality analysis. And that summarizes that pretty quickly.

CHAIR PETTI: What about this tritium drinking water, should we put a sentence in? Or we can put it in on 11 when we get to 11 next April. So they haven't, all they've done is a classic dose calculation, but the tritium drinking water standard may be the more limiting condition that needs to be looked at.

MEMBER HALNON: Okay. Let me think about that because we're dealing with another tritium issue right now that --

CHAIR PETTI: Yes.

MEMBER HALNON: Yes, I'll take to the other, I may draft up a sentence or two and make sure it's consistent so we don't send it in different directions. Any other comments, questions? Steve, thank you for your help on this. Okay, Dave, back to you.

CHAIR PETTI: Okay. I guess we can, once again, go out for public comments. Any member of the public that has a comment, please unmute yourself, state your name, affiliation if applicable, and your comment. Okay, not hearing anything I think we're done.

This is good. We completed in advance of the agenda. And it gives us confidence in terms of the rest of the reviews and now have a sense of a cadence in terms of how much time it's going to take to get through the remaining chapters. So with that we'll recess --

MR. HASTINGS: Mr. Chairman?

CHAIR PETTI: Yes.

MR. HASTINGS: Hi, this is Peter Hastings. I wanted to make just some brief closing remarks on behalf of Kairos.

CHAIR PETTI: Sure.

MR. HASTINGS: So, this is Peter Hastings. I'm the vice president of regulatory affairs for Kairos Power. And I want to thank the ACRS Members and Staff, and the NRC Staff, for their engagement and your detailed review and your comments.

I do want to make a couple of comments myself with respect to some ongoing discussion, both yesterday and today, on the regulatory pathway that we're following. In various comments some members have wondered why there is not more detail than what's available in the PSAR, and have reiterated that Kairos is "proceeding at risk" without regulator assurance over the details of our design.

As you know, we are pursuing a construction permit application under 10 CFR Part 50. And our application was prepared in accordance with the regulation. Primarily 10 CFR 50.34(a). And as further informed by the NRC approved topical report that addresses the differences in regulatory applicability between light water reactors and advance reactors, and also pursuant to the guidance in NUREG-1537, as has been discussed numerous times.

And has also been noted, generally speaking we did not request a finding on safety functions of SSCs for the facility. This is obviously all completely consistent with the regulation. And with the conversations that we've held with the Staff for several years now.

I want to make two things clear. First, to avoid any ambiguity in messaging, the risk of moving forward in a way that the NRC Staff will not find acceptable is solely a commercial programmatic risk, not a safety risk. I'm 100 percent confident that the Members recognize this, but I wanted to make it clear on the record for any outside observers.

Second, as an applicant, Kairos Power appreciates and accepts the programmatic risk of the two-step process under Part 50. We specifically selected this pathway, in part because of the lack of land mines that some of us discovered firsthand on the Part 52 path for first of a kind designs.

We recognized the commercial risk of

moving forward on the details of our design. At the same time I'll observe we've conducted the most comprehensive pre-application engagement of any advance reactor applicant to date. And just recently approved, received approval of the last of our 11 topical reports that apply to, both to our commercial design and to the Hermes Reactor.

And as Chairman Petti remarked earlier, we have significant margins that make it exceedingly unlikely we're painting ourselves into a corner at the construction permit stage. And so for those reasons the commercial risk of moving forward in a way that somehow gets us cross-wise with a regulation and with the Staff is not something that keeps me awake at night. We fully intend and expect to be able to demonstrate all of the regulatory requirements at the OL stage. So thank you again for a productive engagement, and we look forward to the next steps.

CHAIR PETTI: Thank you.

MEMBER DIMITRIJEVIC: Thank you.

CHAIR PETTI: With that we will adjourn the meeting and we will see everyone again April 4th.

(Whereupon, the above-entitled matter went off the record at 2:51 p.m.)

Kairos Power

KP-NRC-2303-001

March 15, 2023

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 Hermes Preliminary Safety Analysis Report Chapters 1, 2, 3, 4, 6, and 9

References: Letter, Kairos Power LLC to Document Control Desk, “Submittal of the Preliminary Safety Analysis Report for the Kairos Power Fluoride Salt-Cooled, High Temperature Non-Power Reactor (Hermes), Revision 2,” February 24, 2023 (ML 23055A673)

This letter transmits the presentation slides for the March 23-24, 2023 briefings for the Advisory Committee for Reactor Safeguards (ACRS), Kairos Power Subcommittee. During the March 23 meeting, participants will discuss Hermes Preliminary Safety Analysis Report (PSAR) Chapters 1, 2, 3, and 4. During the March 24 meeting, participants will discuss PSAR Chapters 4, 6, and 9.

Enclosures 1 and 2 provide the non-proprietary slides for the March 23 and 24 briefings, respectively. 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,

Peter Hastings, PE
Vice President, Regulatory Affairs and Quality

Enclosures:

1) Presentation Slides for the March 23, 2023 ACRS Kairos Power Subcommittee Meeting (Non-Proprietary)
2) Presentation Slides for the March 24, 2023 ACRS Kairos Power Subcommittee Meeting (Non-Proprietary)

xc (w/enclosure):

William Jessup, Chief, NRR Advanced Reactor Licensing Branch
Benjamin Beasley, Project Manager, NRR Advanced Reactor Licensing Branch
Edward Helvenston, Project Manager, NRR Advanced Reactor and Licensing Branch
Samuel Cuadrado de Jesus, Project Manager, NRR Advanced Reactor Licensing Branch
Matthew Hiser, Project Manager, NRR Advanced Reactor Licensing Branch
Weidong Wang, Senior Staff Engineer, Advisory Committee for Reactor Safeguards

Enclosure 1
Presentation Slides for the March 23, 2023
ACRS Kairos Power Subcommittee Meeting
(Non-Proprietary)

Kairos Power

Introduction and Hermes PSAR Chapter 1

DREW PEEBLES – SENIOR LICENSING MANAGER

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 23, 2023

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.

Overview of 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:
    - Over 300 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 2031 (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

Hermes PSAR Overview

- 10 CFR 50 Licensing Pathway
  - Construction Permit Application – Submitted Fall 2021
    - Environmental Report
    - Preliminary Safety Analysis Report (PSAR)
  - Next Licensing Step: Operating License Application
    - Final Safety Analysis Report (FSAR)

- Hermes PSAR Application Format and Content
  - Developed using guidance in NUREG 1537
  - Presents preliminary design and preliminary safety analysis consistent with 10 CFR 50.34(a)
  - PSAR does not request commission approval of the safety of any design feature or specification
    - *10 CFR 50.35(b) A construction permit will constitute an authorization to the applicant to proceed with construction but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit.*

* Not Applicable to Hermes – Chapter has no content

** Minimal Content at PSAR

Hermes PSAR Format

- Chapter 1 – The Facility
- Chapter 2 – Site Characteristics
- Chapter 3 – Design of Structures, Systems, and Components
- Chapter 4 – Reactor Description
- Chapter 5 – Heat Transport System
- Chapter 6 – Engineered Safety Features
- Chapter 7 – Instrumentation and Control Systems
- Chapter 8 – Electric Power Systems
- Chapter 9 – Auxiliary Systems
- Chapter 10 – Experimental Facilities and Utilization*
- Chapter 11 – Radiation Protection Program and Waste Management
- Chapter 12 – Conduct of Operations**
- Chapter 13 – Accident Analysis
- Chapter 14 – Technical Specifications**
- Chapter 15 – Financial Qualifications**
- Chapter 16 – Other License Considerations*
- Chapter 17 – Decommissioning and Possession-only License Amendments*
- Chapter 18 – Highly Enriched to Low Enriched Uranium Conversion*

Kairos Power Reports Referenced in PSAR

- Topical Reports
  - KP-TR-003 – Principal Design Criteria
  - KP-TR-004 – Regulatory Analysis
  - KP-TR-005 – Reactor Coolant
  - KP-TR-007 – Quality Assurance Plan
  - KP-TR-010 – Fuel Performance Methodology
  - KP-TR-011 – Fuel Qualification Methodology
  - KP-TR-012 – Mechanistic Source Term Methodology
  - KP-TR-013 – Metallics Qualification Methodology
  - KP-TR-014 – Graphite Qualification Methodology
- Technical Reports
  - KP-TR-017 – Core Design Methodology
  - KP-TR-018 – Postulated Event Methodology

Hermes PSAR Chapter 1 – The Facility

- The purpose of Hermes is to test and demonstrate the key technologies, design features, and safety functions for KP-FHR technology
  - 35 MWth non-power reactor facility, 4 year licensed lifetime
  - Located in Oak Ridge, Tennessee at the East Tennessee Technology Park (Former site of Oak Ridge Gaseous Diffusion Plant)
- Principal Design Criteria based on NRC-approved topical report, KP-TR-003-NP-A "Principal Design Criteria"
- Low consequences due to inherent safety features
  - TRISO fuel
  - Flibe coolant
- Engineered safety features are provided to contain fission products and passively remove decay heat
- Instrumentation and control system provides monitors and controls plant operations. Electrical System provides the normal and backup power to the facility
- Auxiliary systems include a chemistry control system, inert gas system, tritium management system, fire protection system, heating and cooling systems, etc.

Hermes PSAR Chapter 1 – The Facility

- Nuclear Safety Classifications: Safety-Related or Non-Safety Related
- Potential events are evaluated using a deterministic safety analysis with a Maximum Hypothetical Accident
- Radioactive waste management controls wastes produced by plant operations and radiation protection program protects health and safety of workers
- Experimental capabilities include testing of fuel irradiation, materials corrosion and irradiation, and transient and power maneuvering
    - Capability to perform these activities is included in normal system design described in PSAR
    - No additional facilities or capabilities required
- Research and development programs to resolve safety questions will be resolved before the completion of construction
- Hermes is a single unit reactor that does not share any systems or equipment to perform safety functions

Hermes PSAR 3.1 Introduction and 3.6 Systems and Components

DREW PEEBLES – SENIOR LICENSING MANAGER

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 23, 2023

3.1 Applicable Regulations and Guidance

- Kairos Power is pursuing a construction permit for the Hermes reactor under 10 CFR 50
- The NRC regulations in Title 10 to the CFR were evaluated for applicability and documented in "Regulatory Analysis for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor" topical report (KP-TR-004-NP-A)
- PSAR Table 3.1-1 identifies the design-related regulations that are applicable to the Hermes Test Reactor
  - No specific exemptions from regulations were identified
  - Regulations related to combustible gas control were concluded to be not technically relevant
- Kairos Power evaluated NRC regulatory guides for applicability to the Hermes Test Reactor
  - NRC Division 1 regulatory guides are not applicable to research and test reactors
  - Divisions 2, 4, and 8 apply and were considered for the Hermes Test Reactor, as shown in Section 3.1

3.1 Principal Design Criteria

- Kairos Power has developed a set of Principal Design Criteria (PDC) for KP-FHR technology
- The design criteria were approved in a Topical Report titled "Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor" (TR-003-NP-A)
- These PDCs have been applied to the design of the Hermes Test Reactor, with the following exceptions:
  - PDC 5, Sharing of structures, systems, and components (SSCs) - Satisfied because there is only one reactor and no SSCs are shared with another reactor
  - PDC 73, Reactor coolant system interfaces - Not Applicable to the Hermes Test Reactor because there is no secondary coolant fluid
- The terms “safety-significant,” “anticipated operational occurrences,” and “accidents” used in the PDCs are not applicable to the Hermes reactor and are not used in the PSAR
  - These terms are relevant to power reactors which use frequency to bin postulated events
  - The Hermes safety analysis utilizes a deterministic Maximum Hypothetical Accident (MHA)

3.6 Fundamental Safety Functions

- Prevent uncontrolled release of radionuclides
  - Functional containment (TRISO fuel and Flibe coolant) retains fission products and limits release during normal and postulated events
  - Safety-related fluid systems that may contain circulating radiological activity are designed to ASME Section III
  - Non-safety-related fluid systems that may contain circulating radiological activity are designed to ASME Section VIII, B31.1/B31.3, or applicable API standards
- Remove decay heat in the event of a postulated event
  - Natural circulation and the passive decay heat removal system reject residual heat from the reactor core to the atmosphere
- Control reactivity in the reactor core
  - Reactivity control and shutdown system provides reactivity control during normal and postulated events

3.6 SSC Safety Classification

- SSCs are classified as safety-related or non-safety related
- The 10 CFR 50.2 definition of safety-related for light water reactors is modified for the Hermes Test Reactor as follows:
  - Safety-related structures, systems, and components means those structures, systems, and components that are relied upon to remain functional during and following design basis events to ensure:
    - The integrity of the portions of the reactor coolant ~~pressure~~ boundary relied upon to maintain coolant level above the active core;
    - The capability to shut down the reactor and maintain it in a safe shutdown condition; or
    - The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11
- This departure from 10 CFR 50.2 is necessary because the near atmospheric pressure design and the reactor coolant boundary does not provide a similar pressure-related or fission product retention function as light-water reactors for which these definitions were based
- The classification of SSCs is shown in PSAR Table 3.6-1

3.6 Seismic and Quality Classifications

- Seismic Classification
  - Safety-related SSCs are classified as SDC-3 in accordance with ASCE 43-19
  - Safety-related SSCs are located in the safety-related portion of the reactor building
  - Non-safety-related SSCs are designed to local building codes (ASCE/SEI 7-10)
- Quality Classification
  - Safety-related SSCs are classified as quality-related
  - Non-safety-related SSCs are classified as not quality-related
  - Quality-related SSCs conform to the requirements of the quality assurance program for the Hermes Test Reactor, which is based on ANSI/ANS 15.8
- The seismic and quality classification of SSCs is shown in PSAR Table 3.6-1

Hermes PSAR Chapters 2.1-2.4, 3.2, and 3.3

MARTY BRYAN – SENIOR MANAGER, SITE LICENSING

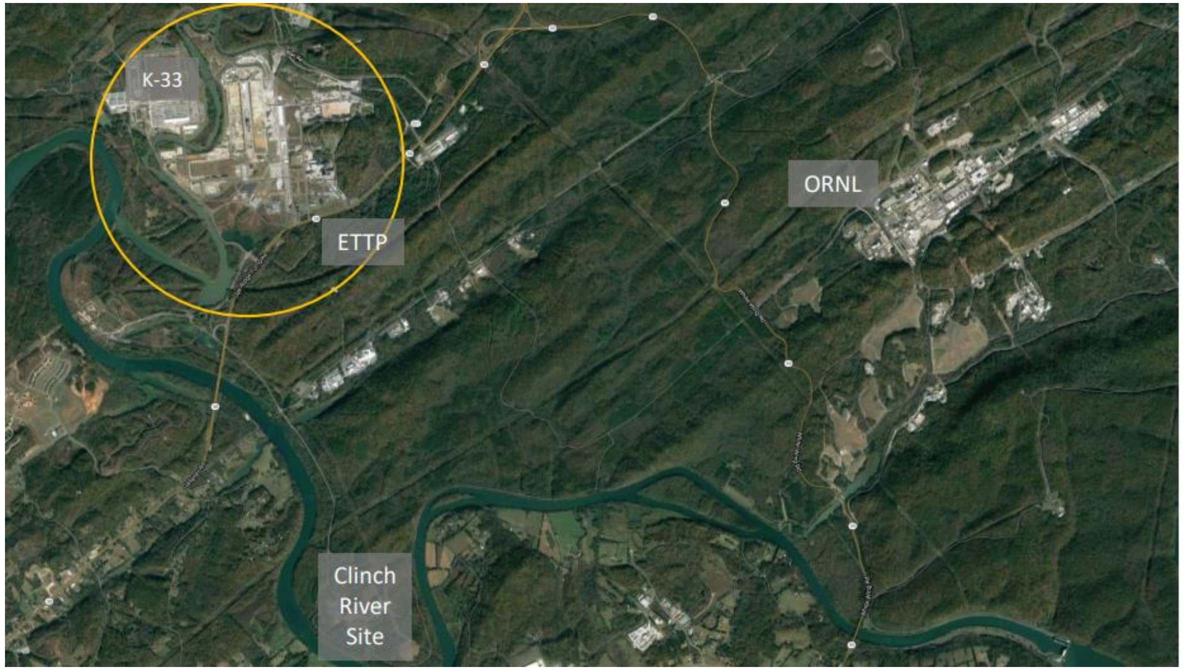
BRIAN SONG – SENIOR MANAGER, CIVIL STRUCTURES

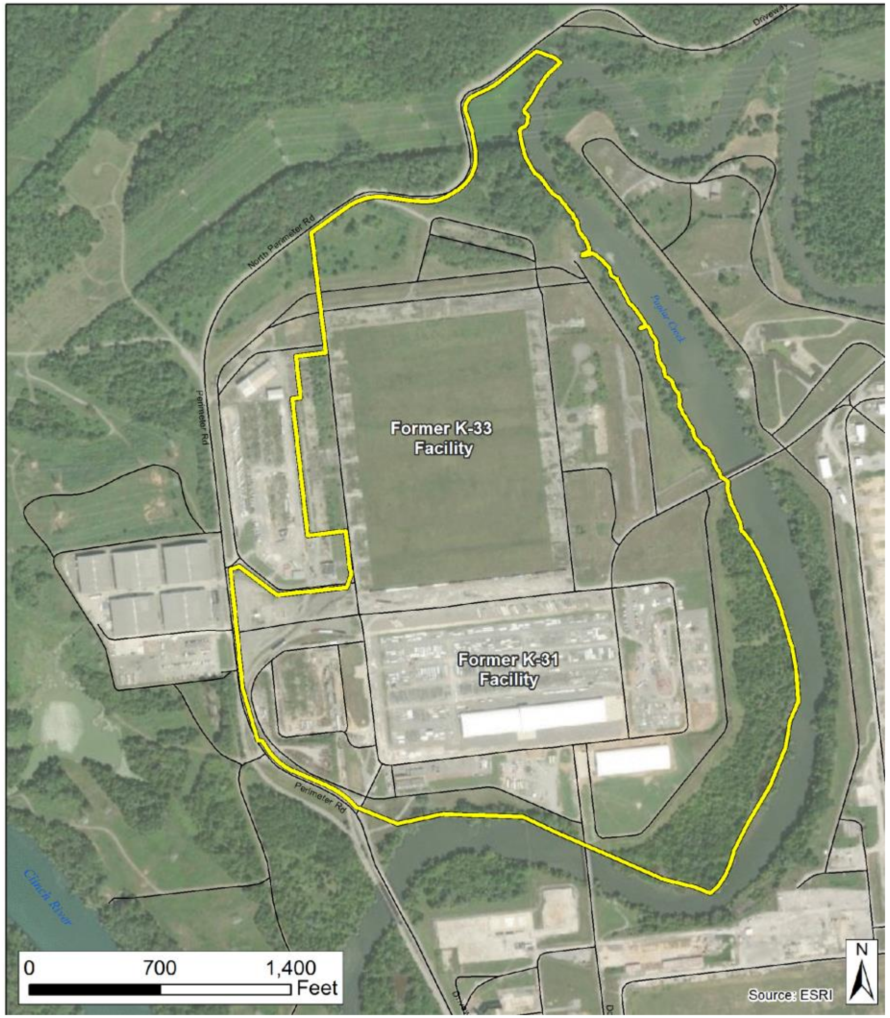
ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 23, 2023

2.1 Geography and Demography: Hermes Site Location

- The site is located in Oak Ridge, Tennessee in Roane County within the East Tennessee Technology Park (ETTP)
- The Hermes test reactor will be located on former Department of Energy gaseous diffusion plant (K-33) building site
- The site boundary encompasses approximately 185 acres
  - About 30 acres would be permanently disturbed for operations of the facility

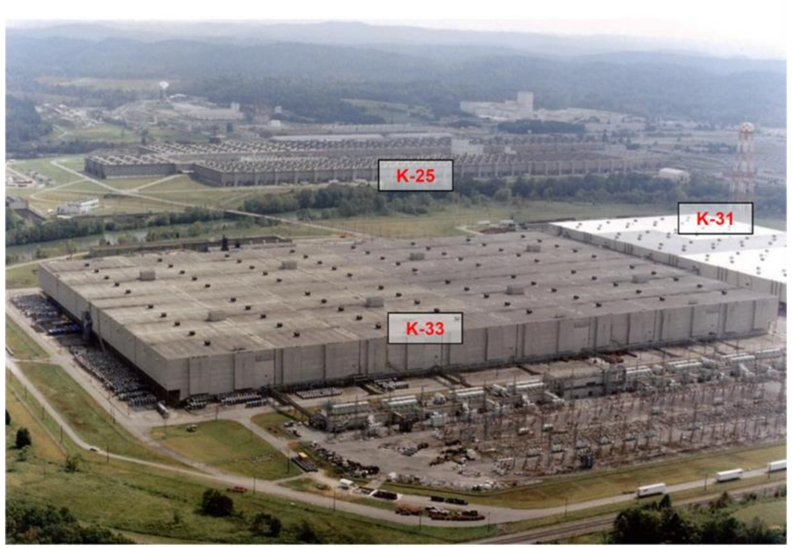




2.1 Geography and Demography: Hermes Site Location

- The original K-33 Building was constructed in 1954
- The uranium enrichment facility ceased operations in 1985
- DOE began reindustrialization of the ETTP in 1996
- The site was released for industrial use in 2011

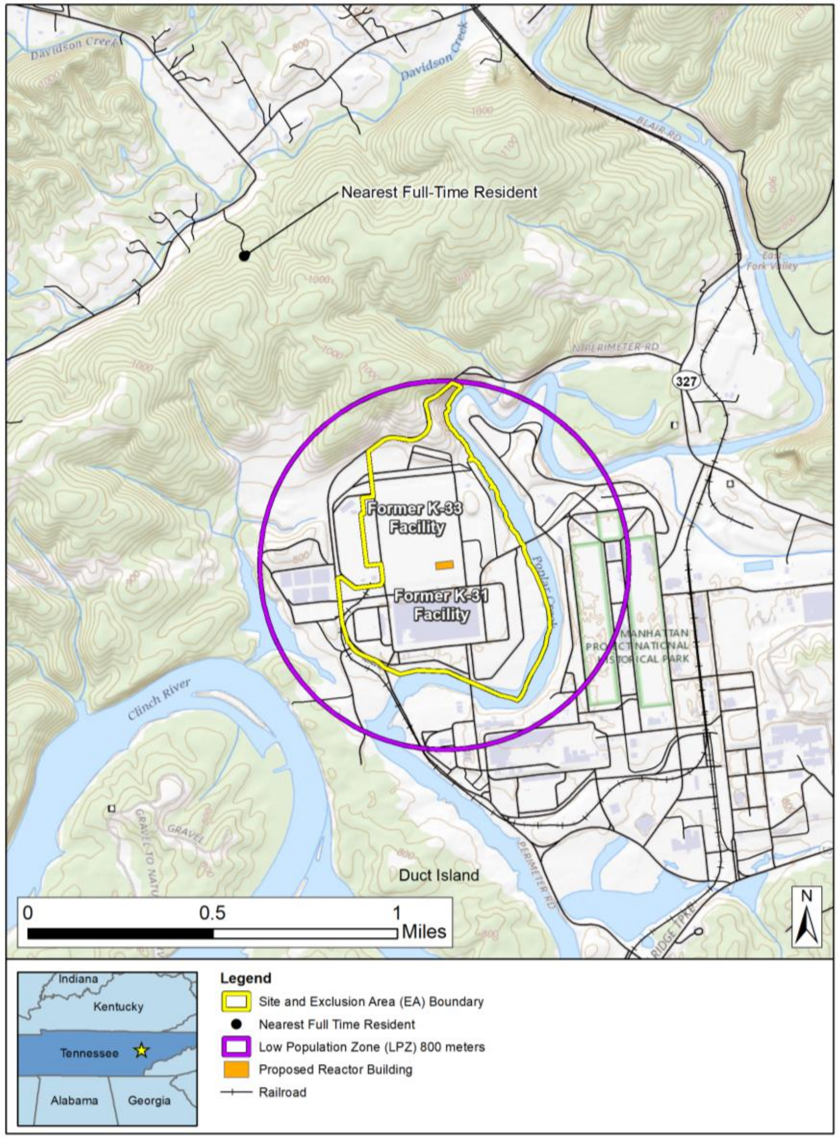
View from West to East



View from North to South

2.1 Geography and Demography: Boundary and Zone Area Maps

- The site boundary is defined by the area owned, leased, or controlled (10 CFR 20.1003)
- The exclusion area boundary is defined as the area within the site boundary where the reactor site management has direct authority over all activities
(10 CFR 100.3 and ANSI/ANS-15.16-2015)
- The low population zone is conservatively set at 800 meters from the reactor
  - The nearest resident is 0.7 mi NW from the site boundary
  - The PSAR includes population data 5 miles from the reactor
- The emergency planning zone is coincident to the site boundary (10 CFR 50, Appendix E.I.3)



2.2 Nearby Industrial, Transportation, and Military Installations

- An investigation of industrial, transportation, and military facilities within 5 miles (8 km) of the site was performed to identify potential external hazards (explosions, flammable vapor clouds [delayed ignition], toxic chemicals, and fires)
- The effects from potential external hazards within 5 miles of the site were determined to not warrant further analysis with the exception of:
  - The distance from the Hermes site to TN-58 was less than the safe distance calculated for shipments of chlorine or anhydrous ammonia. Therefore, the main control room will be designed with detectors for these chemicals.
- There are no existing commercial airports located within 10 miles of the site, however a general aviation airport is proposed to be located less than 1 mile SE of the site
  - The annual probability of an aircraft crashing into the facility was evaluated using the methodology outlined in DOE Standard DOE-STD-3014-2006
    - The total crash frequency for small, non-military aircraft from general aviation or helicopter operations is above the screening acceptance frequency threshold
  - The safety-related portion of the Reactor Building structure will be designed to withstand the impact of a small non-military general aviation aircraft

Chapter 2 and 3 Relationships

Step 1: Define design basis parameter input envelope

Step 2: Define methods to translate inputs into design loads

Step 3: Define protections for safety-related SSCs using design loads

Meteorology
Section 2.3

- Section 3.2

Hydrology
Section 2.4

- Section 3.3

Seismic
Section 2.5

- Section 3.4

Section 3.5

2.3 Meteorology

- The Hermes site is located on a prior U.S. Department of Energy (DOE) nuclear facility site within the DOE-managed Oak Ridge Reservation (ORR)
  - The ORR includes an extensive network of meteorological towers
  - Historical meteorological studies from 1953 and 2011 indicate that basic flow patterns have been in place during the recorded weather history of the ORR area
- Topography influences the weather and climate of the region around the site due to its location between the Cumberland Mountains to the northwest and the Great Smoky Mountains to the southeast.
- Prevailing winds in the region reflect the channeling of airflow from southwest to northeast caused by the orientation of the valleys and ridges

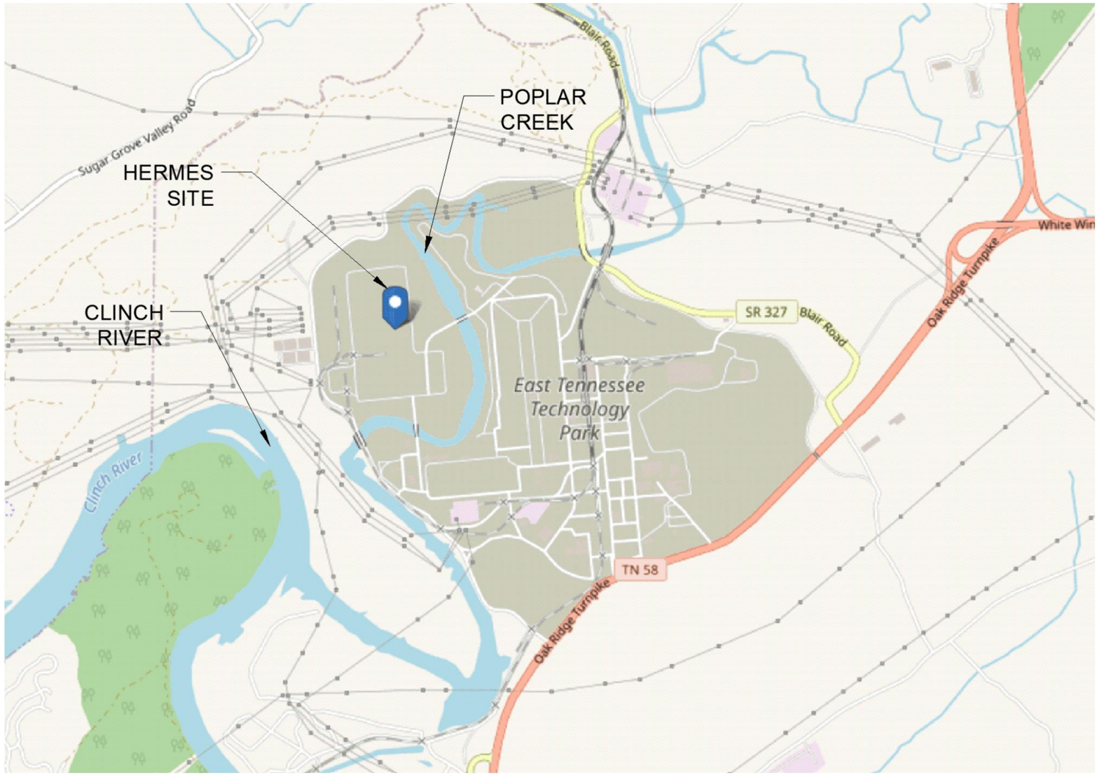
2.3 Meteorology (continued)

- Extreme Winds
  - Estimated extreme winds are based on climatological data from Oak Ridge and Knoxville, Tennessee, and hourly observations from meteorological Tower J (1.1 km southeast of the site) and Tower L (1.6 km southeast of the site)
    - For a 100-year return period, the maximum wind speed is 90 mph
  - Hurricane winds are mainly a concern for coastal locations as shown by the wind speed contours presented in Regulatory Guide 1.221
  - The probability of a tornado occurring at the site is low based on records from the NWS Morristown Tornado Database

- Extreme Precipitation
  - Historical precipitation data for the site were obtained from several surrounding National Weather Service (NWS) and Tennessee Valley Authority (TVA) sites
  - Storms with ice greater than or equal to 1 inch of ice occurred five times in 50 years and storms with ice greater than or equal to 2 inches of ice occurred two times in 50 years
  - The historical maximum snowfall event for a 48-hour period was determined to be 28 inches recorded in Westbourne, Tennessee, from February 19, 1960 to February 21, 1960

2.4 Hydrology: Description

- The site is located near the confluence between Clinch River and Poplar Creek
  - TVA manages water levels year-round for dam safety and flood control
  - Both Clinch River and Poplar Creek are considered as potential flooding sources
- The grade level for the site is 765 feet above mean sea level (feet msl)
  - The normal water surface elevation for Poplar Creek near the site is 744 feet msl (21 feet lower than site grade)



2.4 Hydrology Characterization: Previous Flood Studies

- There are two previous flood studies with estimated flooding elevations in the vicinity of the ETTP Hermes site:
  - FEMA Flood Insurance Study for Roane County, TN
    - Includes 10-, 50-, 100-, and 500-year return periods
    - All flood elevations from this study are below the Hermes site grade of 765 feet above mean sea level (feet msl)
  - Flood Hazard Evaluation for UCOR dated April 2015
    - A large range of return period floods (25 year to 100,000 year) were modeled and estimated
    - Results were assessed and used to identify a preliminary design-basis flood

2.4 Hydrology: Credible Hydrology Events and Design Basis

- The credible hydrological event for the Hermes site is selected as a 25,000-year return period (exceedance probability of 4E-5), consistent with Flooding Design Category 4 (FDC-4)
  - This results in a design basis flooding level for the site at 759.9 feet msl, based on previous studies
    - 5.1 feet below plant grade of 765.0 feet msl
- The Hermes site layout and grading plan takes advantage of the existing site topography so that storm water runoff naturally drains to the east, south, and west with flow directed to Poplar Creek

3.2 Meteorological Damage

- The design of SSCs considers the potential for meteorological damage, including rain, snow, wind, tornado, and tornado and wind-borne missiles for the site
- The safety-related portion of the reactor building structure provides protection to safety-related systems and components from meteorological damage
  - No credit is taken for the non-safety-related portions of the reactor building (exterior shell)
- Design basis meteorological parameters applicable to the design of the safety-related portion of the reactor building structure are established for: normal wind loads, high wind loads (tornados and hurricanes), and precipitation loads
- Normal wind load design basis:
  - Local building codes cite ASCE/SEI 7-10, “Minimum Design Loads for Buildings and Other Structures”. This standard defines risk categories for structures and includes design basis normal wind velocities for each risk category.
  - Risk Category IV (for hazardous substances) is the most stringent and selected as the design basis for the safety-related portions of the Reactor Building
  - Risk Category IV results in a design basis wind velocity of 120 miles per hour (mph)
    - This wind velocity bounds the site characterization meteorological data
    - This is based on a 1700-year mean recurrence interval, which is more conservative than the 100-year return period
  - The applied normal wind loads are determined using ASCE/SEI 7-10 Risk Category IV and exposure category C

3.2 Meteorological Damage (continued)

- High wind load design basis:
  - Guidance from Regulatory Guide (RG) 1.76, Revision 1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," was used to determine characteristics of the design-basis tornado
    - The applied tornado wind loads are determined using the methods in ASCE/SEI 7-10 and the wind speeds from RG 1.76 for Region I
    - The loads from tornado-generated missile impacts are transformed into an effective or equivalent static load consistent with NUREG-0800, Section 3.5.3, Subsection II using the missile spectrum and maximum horizontal speeds provided in Table 2 of RG 1.76 for Region I
  - Guidance from RG 1.221, Revision 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," was used to determine applicable design parameters for hurricane loads
    - The applied hurricane wind loads are determined using the methods in ASCE/SEI 7-10 with a maximum wind speed of 130 mph and velocity pressure based on the guidance in RG 1.221 for the site location
    - The loads from hurricane-generated missile impacts are transformed into an effective or equivalent static load consistent with NUREG-0800, Section 3.5.3, Subsection II using the missile spectrum from RG 1.221

3.2 Meteorological Damage (continued)

- Precipitation load design basis:
  - The grading and drainage design for the site will preclude loads from precipitation accumulation on the ground affecting the safety-related portion of the Reactor Building
  - The non-safety related exterior shell of the Reactor Building has a sloped roof, therefore, loads due to rain accumulation are not considered as a structural load in the structural design.
  - Similarly, as a result of the lack of rain accumulation, load due to ice is anticipated to be minimal and is therefore enveloped by the snow load
  - The snow load design parameters are based on Chapters 1 and 7 of ASCE/SEI 7-10 for Risk Category IV structures and site location
  - The applied structural snow loads are determined based on the ground snow load of 21.9 psf and using the methods in ASCE/SEI 7-10 for Risk Category IV structures
    - Load considerations include balanced snow loads, unbalanced snow loads, snow drift loads, and rain on snow surcharge loads

3.3 Water Damage

- The design of the safety-related portions of the reactor building considers the loads from both external and internal flooding events
- External flooding postulated events do not pose a hydrologic load because the grade elevation is above the design basis flood elevation determined in PSAR Section 2.4
- Internal flooding postulated events consider the water sources within the safety-related portions of the reactor building
  - As discussed in Section 3.5, safety-related SSCs are protected from internal flooding:
    - Safety-related SSCs vulnerable to flooding are elevated, shielded or otherwise protected from spray. This includes Flibe-bearing components.
    - Design features direct water flow and prevent it from entering enclosures containing safety-related SSCs.
    - The volume of water in the safety-related portions of the reactor building is limited by design. For water systems that cross the base isolation moat, automatic or manual termination of flow will be specified in the operating license application.

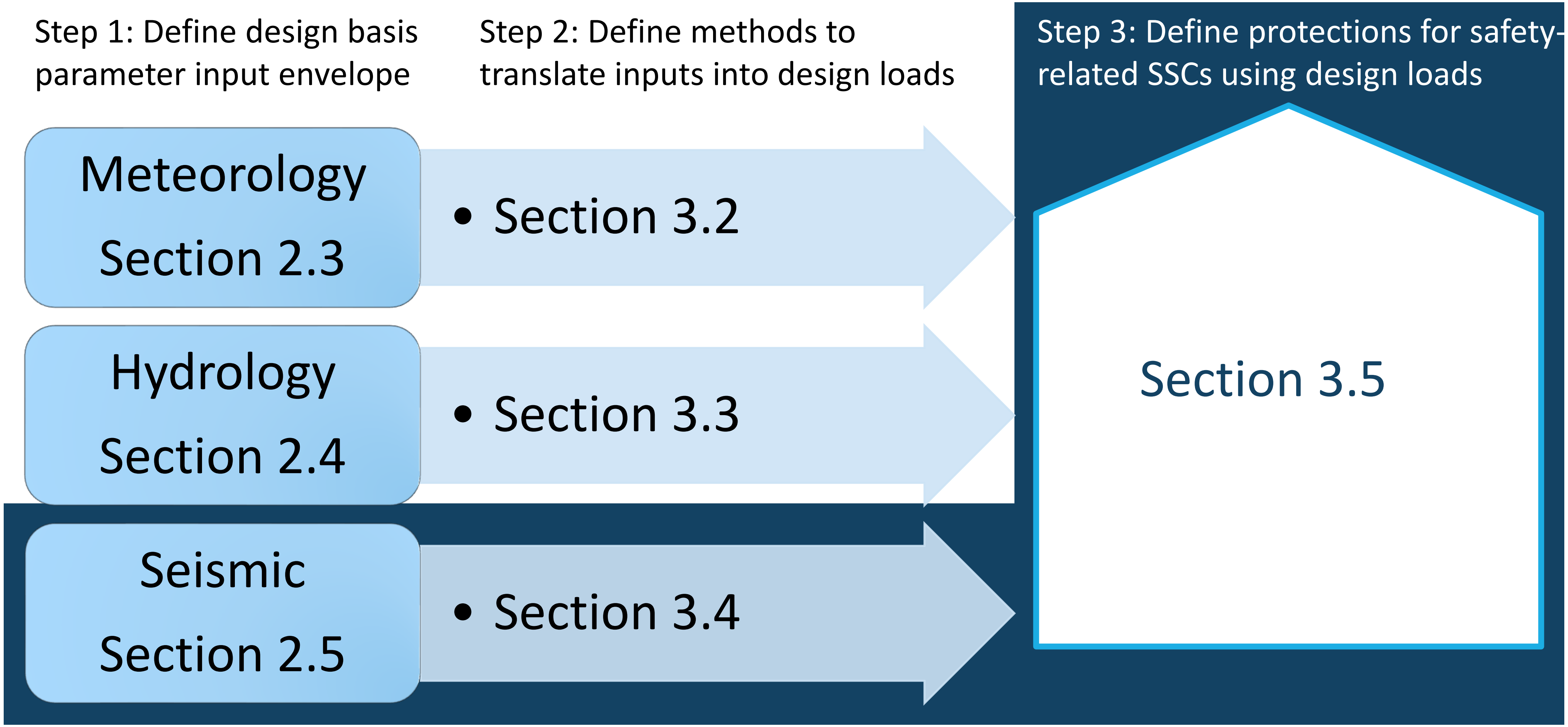
Hermes PSAR 2.5, 3.4, and 3.5
Geology, Seismic Design, and Reactor Building Structures

BRIAN SONG – SENIOR MANAGER, CIVIL STRUCTURES

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 23, 2023

Chapter 2 and 3 Relationships



2.5 Geology, Seismology, and Geotechnical Engineering

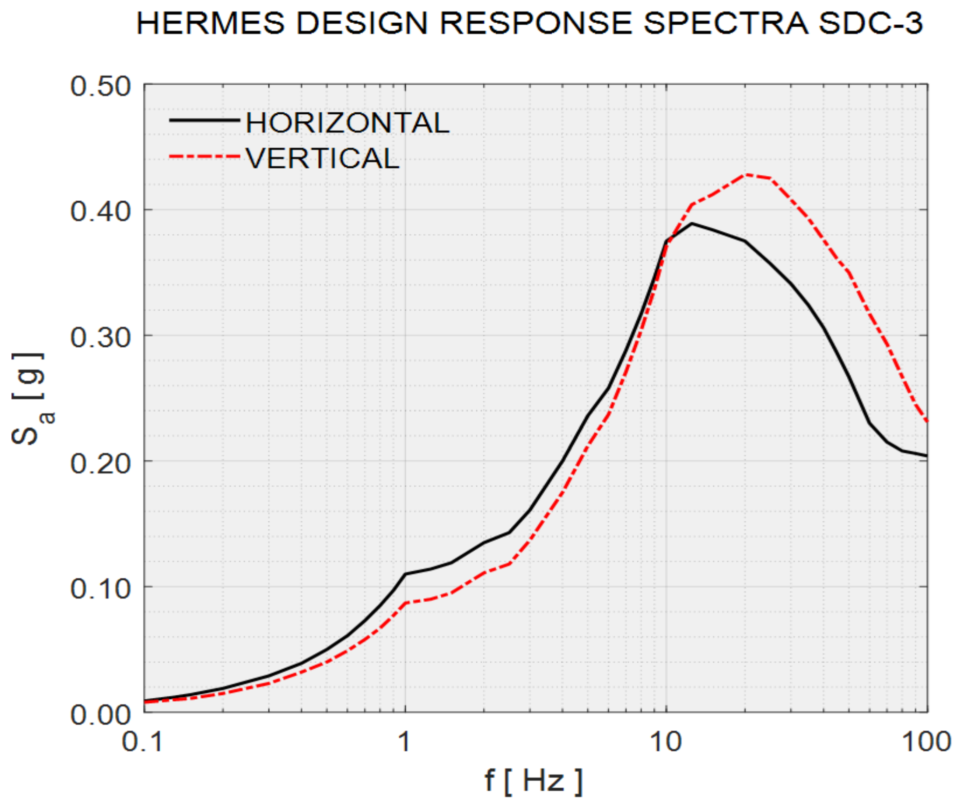
- Section 2.5 characterizes the geologic, geophysical, seismic and geotechnical aspects of the region and site to develop a seismic design basis for the facility
- The Hermes PSAR relies on existing information from the Clinch River Early Site Permit Application (CR-ESPA) for the regional and local geologic description, with supplemental information as needed
  - Covers 200 miles around the site
  - The CRNS site is close (3.5 miles) to the Hermes site and shares the same regional geology
- The Hermes Probabilistic Seismic Hazard Analysis (PSHA) is adapted from the CR-ESPA PSHA supplemented with consideration of current seismic hazard publications for the site and regional area
  - The PSHA methodology is an enhancement over the guidance in NUREG 1537
  - The CRNS PSHA meets ANSI/ANS 2.29 "Probabilistic Seismic Hazard Analysis”

2.5 Geology, Seismology, and Geotechnical Engineering: Site Geology

- CRN site geology information is directly applicable to the Hermes site
- A subsurface stratigraphy was developed for the Hermes site from a geotechnical boring program
- The placement of the facility on the site was informed by the geotechnical information

2.5 Geology, Seismology, and Geotechnical Engineering: Vibratory Ground Motion Analysis

- Uses CRN PSHA to develop the Seismic Design Response Spectra (DRS)
- Analysis relies on information from the CR-ESPA, with supplements
    - Use of the CR-ESPA, PSHA is both appropriate and reasonable given the proximity between both sites
- The Seismic Source Characterization is based on the CEUS (Central and Eastern United States) Seismic Source Characterization report
- The DRS meets ASCE 43-19 and uses Seismic Design Category 3 for safety-related SSCs which is appropriate for a non-power reactor application



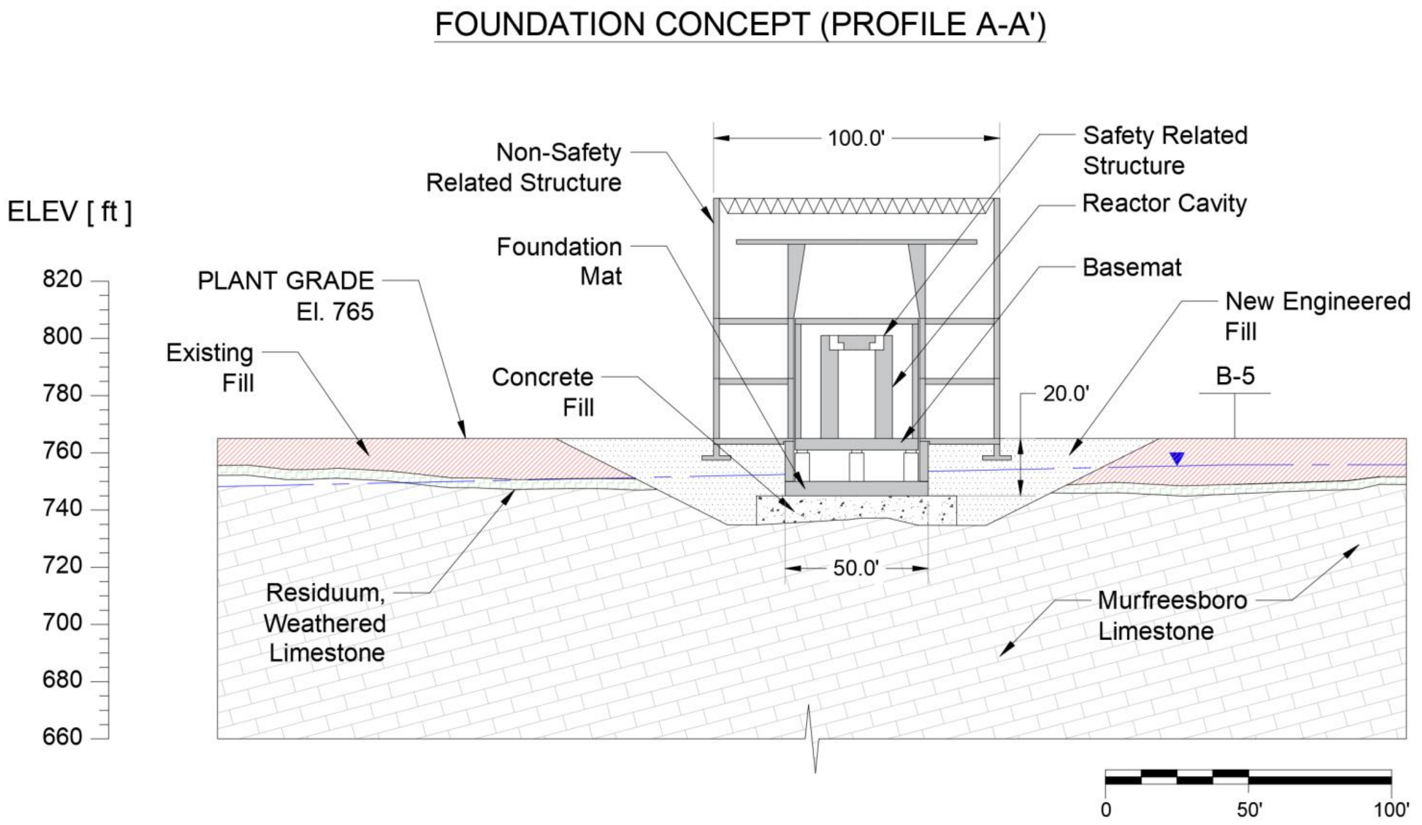
SDC-3 Performance Goal: 1E-4

2.5 Geology, Seismology, and Geotechnical Engineering: Subsurface Deformation

- Relies on information from the CR-ESPA, supplemented by site-specific assessments to assess the potential for sinkholes, faults, and/or soil liquefaction
  - Given the subsurface conditions, and foundation interface plans along with fill placement, there is no potential for liquefaction at the site
  - Only inactive surface faults have been documented within the site area
  - The foundation rock for the Hermes reactor is at depths at which no evidence of karstic dissolution is encountered

2.5 Geology, Seismology, and Geotechnical Engineering: Foundation Interface

- The foundation layout has been established based on knowledge of the site subsurface conditions gathered from both historical documentation, including the CR-ESPA, and the subsurface boring exploration campaign
- The bearing system for the safety-related structure is a foundation mat resting on concrete fill over the Murfreesboro rock



3.4 Seismic Damage

- The graded performance-based approach from ASCE 43-19 is used to design the protections for safety-related SSCs from design basis earthquakes
  - Safety-related SSCs are designed to SDC-3, non-safety related SSCs are designed to local building code, which is consistent with NUREG-1537, IAEA-TECHDOC-403, and IAEA-TECHDOC-1347
  - The return period associated with the design basis ground motion corresponding to SDC-3 is similar to the maximum earthquake specified in building codes with a 2% probability of exceedance in 50 years
    - Consistent with NRC approvals for other non-power reactors
    - Additional margin exists due to the short operating lifetime of Hermes
- Seismic performance criteria are consistent with ANSI/ANS 15.7, Research Reactor Site Evaluation
- The 5% damped horizontal and vertical design response spectra are developed consistent with ANS 2.29, using the DRS defined in Section 2.5
- Structural design of non-safety related SSCs is performed in accordance with the 2012 International Building Code and the Tennessee Building Code

3.4 Seismic Damage: Analysis Models

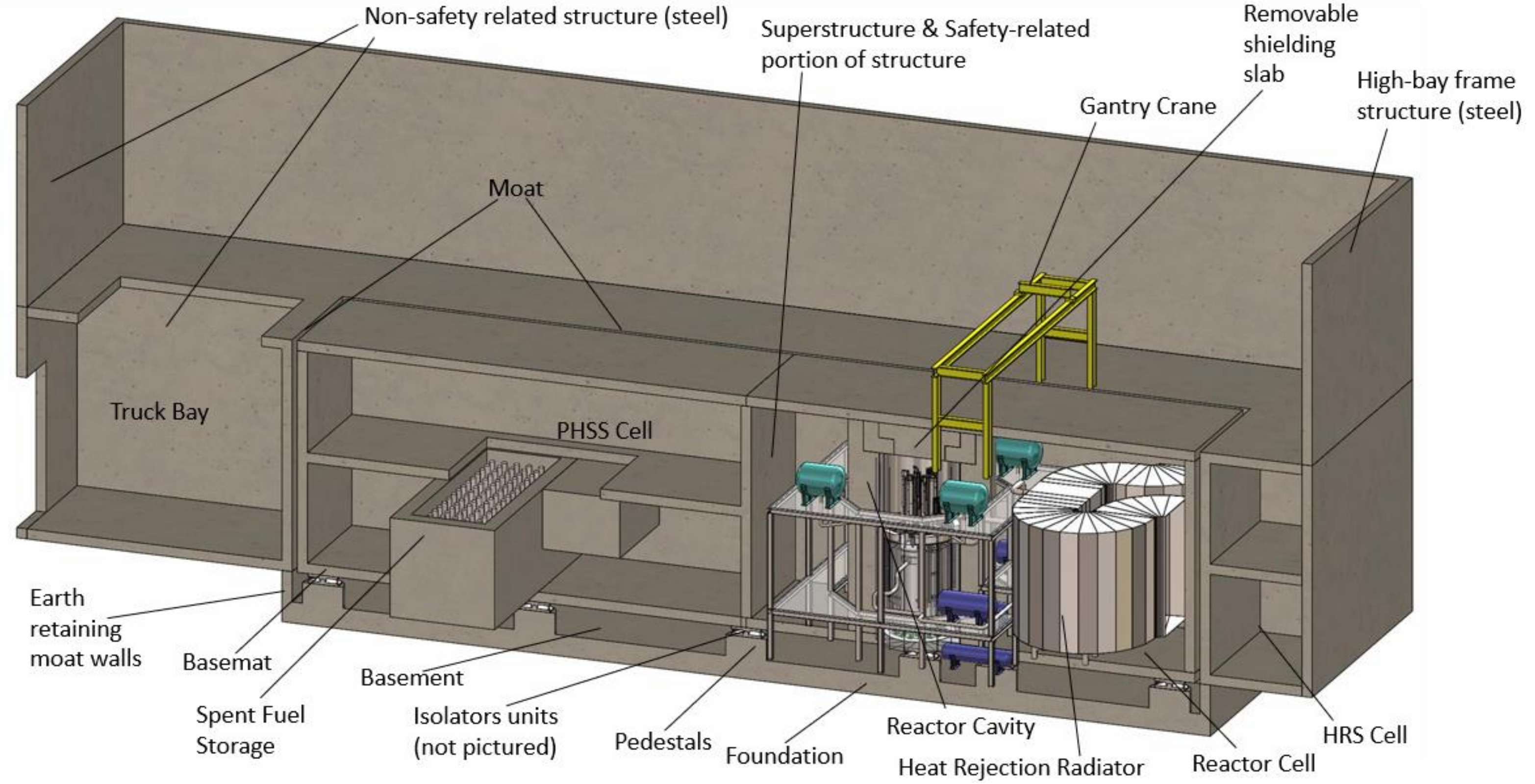
- A 3-D finite element model of safety-related structures will be used for seismic analysis consistent with ASCE 4-16
  - Cracking analysis applies ASCE 4-16 Table 3-2
  - Structural damping applies ASCE 4-16 Table 3-1
  - Structural mass captures self-weight of structural elements as well as portions of design live loads and design uniform snow load
- Models use 3-component seismic input to develop structural forces and in-structure response spectra. Used for SDC-3 structural and equipment qualification.
- Seismic response analysis meets ASCE 4-16, Chapter 4, using deterministic, linear analysis
- Soil-structure analysis will be consistent with ASCE 4-16, Chapter 5

3.4 Seismic Damage: Seismic Instrumentation

- Seismic instrumentation will be installed for monitoring seismic events
- Tri-axial time-history accelerometers will be located in the free field and in the safety-related portion of the reactor building

3.5 Plant Structures: Reactor Building

- ~200’ long, 100’ wide
- Sloped roof
- The safety-related portion of the building uses base isolation using spring/dashpot elements
  - Reactor Cell: vessel, Flibe inventory, and HRR
  - Fuel Cell: PHSS, spent fuel storage
- No other building on the site performs a safety function, including the building that houses the main control room



3.5 Plant Structures: Reactor Building (continued)

The safety functions of the safety-related portion of the Reactor Building are:

- Protection of safety-related SSCs from design basis natural phenomena and external hazards
- Structural support for safety-related SSCs located on the safety-related portion of the Reactor Building
- Protection from adverse effects of non-safety related SSCs failures on the ability of safety-related SSCs to perform their safety functions
- Prevent interactions between reactor coolant (Flibe) and water contained in concrete in the safety-related portion of the reactor building

Note: No part of the reactor building is credited to meet the functional containment safety function

3.5 Plant Structures: Reactor Building Design Criteria

PDC	Description
1	Designed using consensus standards and in accordance with the applicable quality assurance program (ASCE/SEI 7-10).
2	Protects safety-related SSCs from the effects of design basis meteorological, flooding, and seismic events (see Slide 14 for seismic events).
3	Design minimizes the probability and the effect of fires and explosions. (Use of low-combustible materials, separation, fire protection program.)
75	Design protects the geometry of the decay heat removal system from postulated natural phenomena events. (DHRS is located in the safety-related portion of the Reactor Building.)
76	Design permits periodic inspection and surveillance of safety-related structural areas (to be demonstrated in the final safety analysis report).

PDC 2: Seismic Events

- The safety-related portion of the reactor building is a reinforced concrete structure designed to meet ACI 349-2013, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary." Internal steel structures are designed to meet AISC N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities."
- By meeting ASCE 43-19, the safety-related portion of the building provides protection to safety-related SSCs from design basis earthquakes
  - Seismic acceptance is checked for both strength- and displacement-based criteria
  - Limit states are set based on the target performance goals
- Safety-related portion of the Reactor Building uses a spring/dashpot seismic isolation system, which lowers seismic demands on safety-related reactor building and safety-related SSCs in both horizontal and vertical directions
  - The moat is sized to accommodate a displacement consistent with the isolation system meeting the performance goal of 1E-4 per year
  - Design features accommodate potential differential displacements for SSCs that cross the moat

Hermes PSAR 4.2 Reactor Core

RYAN LATTA – PRINCIPAL ENGINEER, FUELS AND MATERIAL

ODED DORON – SENIOR DIRECTOR, REACTOR SYSTEMS DESIGN

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 23, 2023

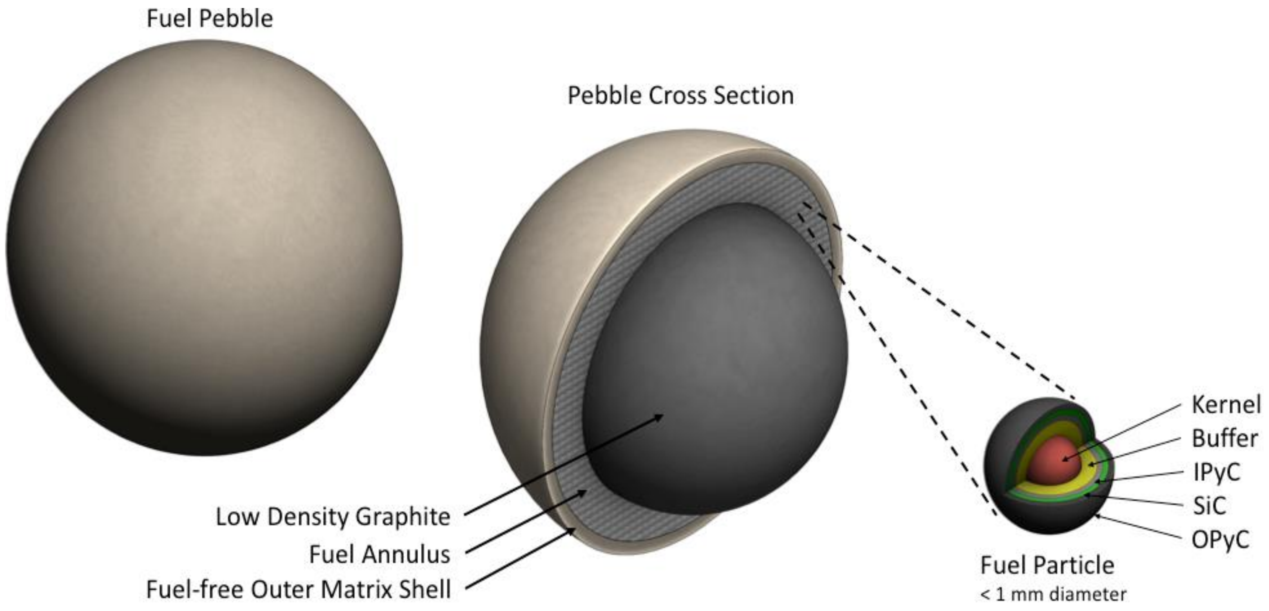
4.2.1 Reactor Fuel

RYAN LATTA – PRINCIPAL ENGINEER, FUELS AND MATERIAL

4.2.1 Reactor Fuel: Fuel Description

- Hermes Test Reactor uses tri-structural isotropic (TRISO) fuel particles in a pebble-based fuel form
- TRISO particle fuel specification is equivalent to the DOE Advanced Gas Reactor (AGR) program
- The kernel and multiple layers of the TRISO fuel particle constitute a primary portion of the functional containment
- Hermes fuel pebble design consists of three regions:
  - Low-density carbon matrix inner core
  - Fuel annulus with TRISO-coated fuel particles embedded in a carbon matrix
  - Fuel-free carbon matrix outer shell
- Moderator pebbles are homogeneous carbon matrix pebbles that do not contain fuel
  - The mixture (ratio) of fuel and moderator pebbles is designed for optimal moderation in Hermes

Annular Fuel Pebble



4.0-cm diameter, annular fuel pebble is about the same size as a golf ball

4.2.1 Reactor Fuel: Fuel Description

Fuel Particle Description

Property	Nominal Value
Kernel diameter (μm)	425
Buffer thickness (μm)	100
PyC thickness (μm)	40
SiC thickness (μm)	35
Kernel density (g/cm^3)	$\geq$ 10.4
Buffer density (g/cm^3)	1.05
PyC density (g/cm^3)	1.90
SiC density (g/cm^3)	$\geq$ 3.19

Fuel Pebble Description

Property	Nominal Value
Pebble radius (cm)	2.0
Overall density (g/cm^3)	1.74
TRISO particles packing fraction	~37%
Pebble uranium loading (g)	6.0
Number of particles per pebble	~16,000

4.2.1 Reactor Fuel: Fuel Qualification

- The Hermes fuel qualification approach is described in topical report KP-TR-011-P "Fuel Qualification Methodology for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor"
- The Hermes TRISO particle fuel specification is equivalent to the DOE AGR fuel specification
  - The EPRI TRISO topical report (EPRI-AR1(NP)-A) demonstrated that the AGR-2 irradiation test resulted in low failure fractions in particles manufactured and inspected to meet the fuel specification
- A PIRT was conducted to evaluate fuel particle and pebble phenomena against a figure of merit
  - The results of the PIRT informs the fuel qualification program
- Pebble laboratory testing in the fuel qualification program demonstrates reasonable assurance the annular pebble will meet functional requirements
  - Mechanical tests – structural integrity
  - Tribology in molten salt and inert gas environments – wear
  - Molten salt infiltration tests – buoyancy
  - Material compatibility tests in salt and air environments – material interaction

4.2.1 Reactor Fuel: Fuel Qualification Envelope

- The Hermes fuel operating envelope is bounded by the fuel qualification envelope established in the fuel qualification methodology topical report
  - The fuel qualification envelope is based on the DOE AGR-2 irradiation and safety tests

Parameter	**TRISO Particle Qualification Envelope**
Peak SiC Layer Temperature – Normal Operation (°C)	1360
Peak SiC Layer Temperature - Transient (°C)	1600
Burnup (%FIMA)	13.2
Peak Particle Power (mW)	155
Peak Fluence ($x10^{25} n/m^2$, E>0.1MeV)	3.8

4.2.1 Reactor Fuel: Fuel Surveillance

- The inert cover gas and Flibe coolant activity levels are monitored to detect an increase in fuel particle failure
- Fuel pebbles are examined in the pebble handling and storage system (PHSS) after exiting the core
  - Pebbles are examined for gross damage – wear, cracking, missing surfaces
  - Burnup is measured to confirm it is less than the qualification envelope, allowing pebble recirculation
- Pebbles near the design burnup limit and those exhibiting indications of damage are removed from service and placed in storage

4.2.1 Reactor Fuel: Fuel Design Bases

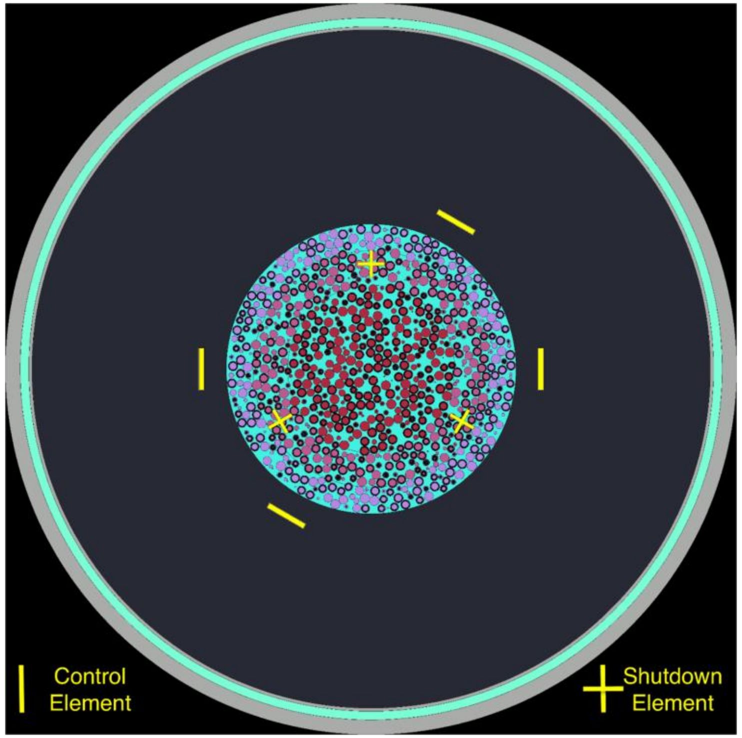
- The fuel is designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits (SARRDLs) are not exceeded (PDC 10)
  - The annular fuel pebble design improves heat transfer by locating TRISO particles near the coolant allowing high operating powers while remaining within temperature limits
  - The TRISO fuel particle design has an equivalent fuel manufacturing specification as the AGR program
  - Fuel particles operate within the qualification envelope that is based on the AGR-2 irradiation and safety tests
- The fuel particle is designed with multiple barriers to constitute the primary portion of the functional containment which controls the release of radioactivity to the environment (PDC 16)
  - The TRISO fuel particle contains four barriers to the release of radionuclides
  - Pebble inspection in the PHSS ensures pebbles operate within the qualification envelope and are not damaged
  - Pebble laboratory testing confirmations that pebbles meet functional requirements, protecting the TRISO particles from damage

4.2.2 Reactivity Control and Shutdown System

ODED DORON – SENIOR DIRECTOR, REACTOR SYSTEMS DESIGN

4.2.2 Reactivity Control and Shutdown System

- Reactivity Shutdown System (RSS)
  - Credited for reactor trip and shutdown
  - 3 safety-related shutdown elements that insert directly into pebble bed
- Reactivity Control System (RCS)
  - Inserted on reactor trip, but not credited
  - 4 non-safety-related control elements that insert into reflector
- Release Mechanism
  - Safety-related electromagnetic clutch
- Drive Mechanism
  - Non-safety-related motor-driven sheave to position element
  - Provides for position indication
- Testing and Inspection
  - RCSS periodically inspected for wear
  - Reactor coolant periodically sampled for an increase in boron concentration that could indicate shutdown element cladding failure
  - RCSS elements can be replaced if necessary



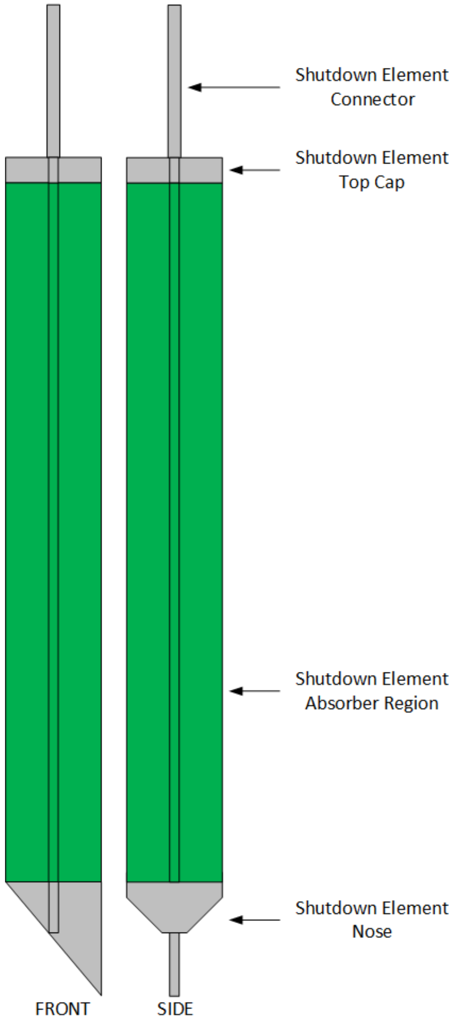
Hermes Core Layout

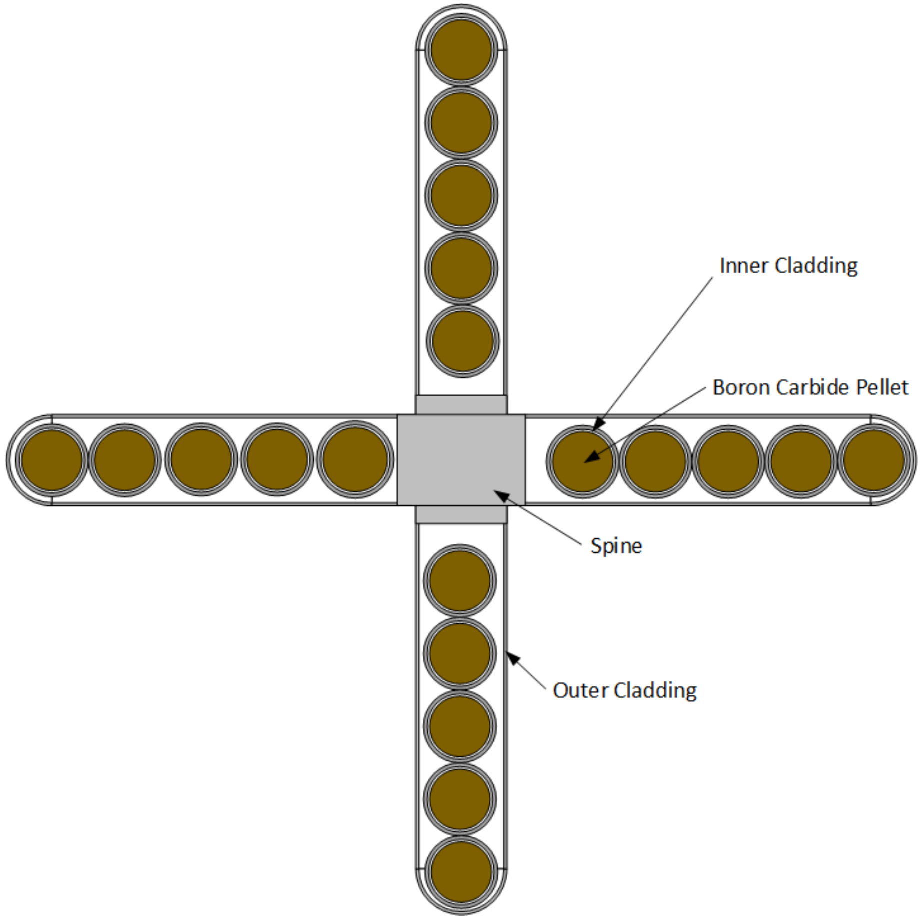
3 in-bed shutdown elements

4 ex-core control elements

4.2.2 Reactivity Control and Shutdown System: Shutdown Elements

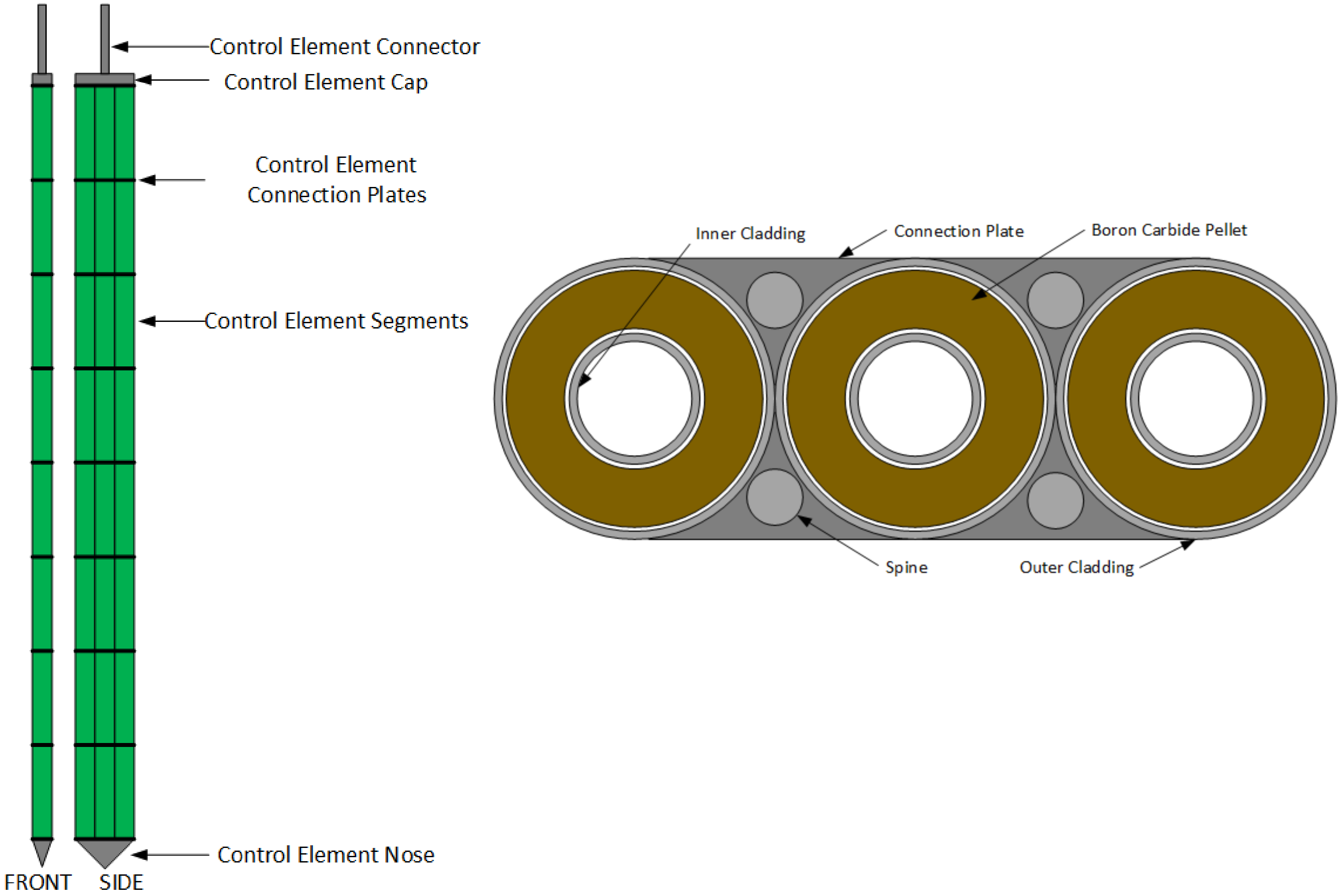
- Shutdown Element
  - Cruciform Design
  - Inner Cladding contains absorber
  - Argon fill
  - Absorber: B_4C
  - Cladding: SS-316H





4.2.2 Reactivity Control and Shutdown System: Control Elements

- Control Element
  - Segmented Annular Design
  - Individual Capsules
  - Argon fill
  - Absorber: B_4C
  - Cladding: SS-316H



4.2.2 Reactivity Control and Shutdown System: Design Bases

- Safety-related RSS is capable of operating during an earthquake. Insertion capability confirmed via testing with maximum deflection of insertion path due to an earthquake. (PDC 2)
- RSS is compatible with environmental conditions and confirmed by qualification testing. Analysis demonstrates internal gas pressure due to irradiation does not exceed safety-related RSS element clad stress limits. (PDC 4)
- RSS is designed to fail in a safe state when the plant trips or upon loss of normal power. The energy holding relays close to remove power supply holding shutdown elements in place and a loss of power allows shutdown elements to drop via gravity. (PDC 23)
- The RCSS (RCS and RSS) meets PDC 26 (discussed in Section 4.5, Nuclear Design)
- RCSS (RCS and RSS) is designed to limit the amount and rate of reactivity insertion by controlling the maximum withdrawal speed of control and shutdown elements (PDC 28)
- The design of the RSS trip function in conjunction with the reactor protection system assures an extremely high probability of accomplishing its safety-related function. Both the RSS and the RCS provide significant negative reactivity insertion into the core via gravity and motor driven means upon a reactor trip. (PDC 29)

Hermes PSAR 4.5 Nuclear Design

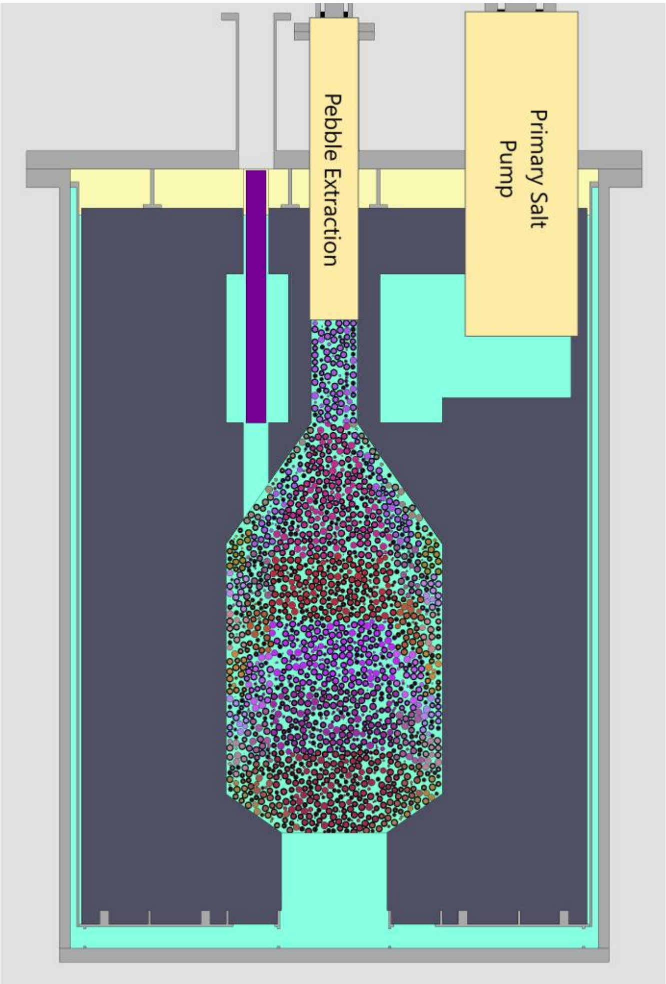
NADER SATVAT – SENIOR MANAGER, REACTOR CORE DESIGN

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 23,2023

4.5 Nuclear Design

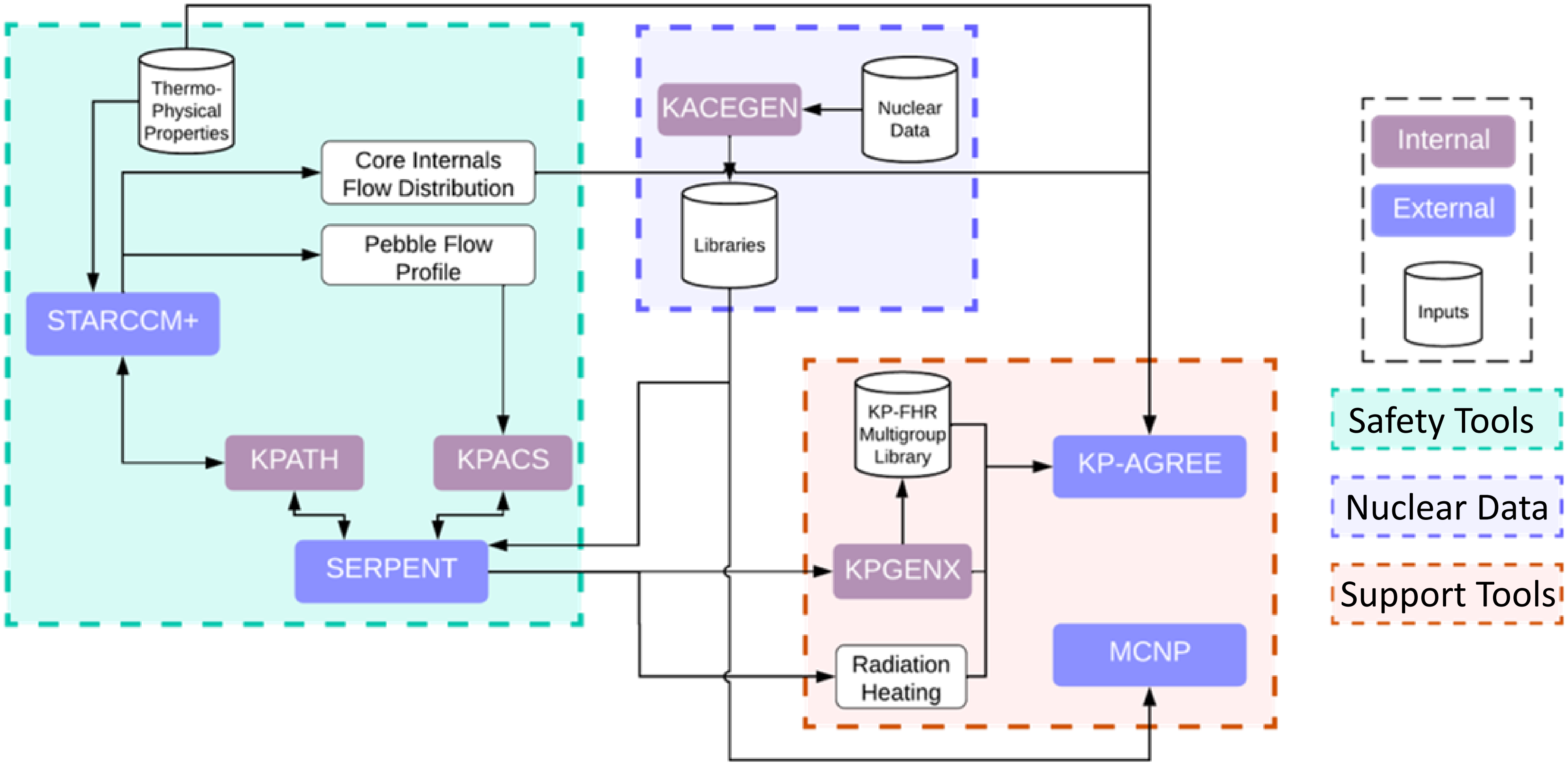
- Reactor core is a packed bed with spherical pebbles
  - Fuel pebbles contain ~6 grams of uranium
  - Fuel pebbles have enrichment up to 20 wt% U-235
  - Moderator pebbles used to improve neutron moderation
  - Core contains approximately 60% pebbles (fuel and moderator) and 40% reactor coolant by volume
  - Core is under-moderated (negative temperature and void feedback)
- Reactor core is continuously refueled
  - Both fuel and moderator pebbles are introduced into the core from the bottom by the pebble handling and storage system (PHSS) and slowly move to the top in ~30-50 days and removed from the core by the PHSS
  - Pebbles inspected for physical damage and burnup
  - Pebbles discharged as they approach their design burnup
- Reactor core is surrounded by a graphite reflector
  - Increases neutron economy, provides moderation/reflection, shields the reactor structures, and maintains the core geometry
- Core design methodology described in "KP-FHR Core Design and Analysis Methodology" (KP-TR-017)



4.5 Nuclear Design

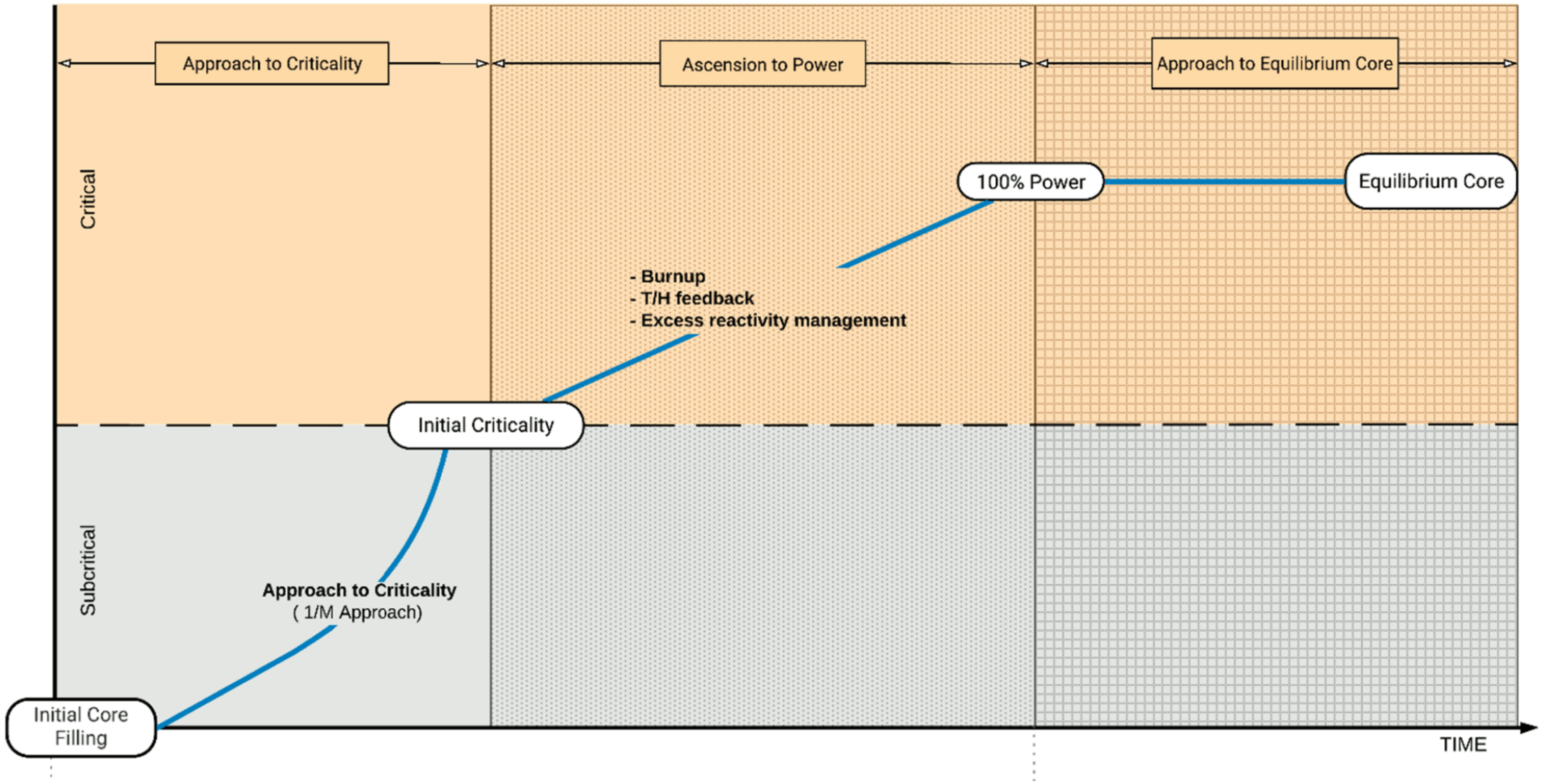
Power	35 MW_{th}
Method for Calculation	Serpent 2 (neutronics); STAR-CCM+ (DEM and T/H)
Coolant	Flibe
Shutdown margin	$k_{eff} < 0.99$
Reactivity Control Elements	7 total; 3 shutdown elements, 4 control elements
Vessel Irradiation	< 0.1 dpa
Reactor Inlet Temperature	550°C
Max Core Outlet Temperature	650°C
Core Volume	2.0 m^3
Enrichment	< 20 wt% U-235
Reactivity Coefficients	Net negative reactivity coefficient; under-moderated

4.5 Nuclear Design: Analytical Methods



4.5 Nuclear Design: Core Life Cycle

- Four cycles of life of the core:
  - Startup and approach to criticality
  - Power ascension
  - Transition to equilibrium (initial power plateau)
  - Equilibrium



4.5 Nuclear Design: Core Operational Regimes

- Approach to criticality
  - A combination of fresh fuel, natural uranium, and moderator pebbles are added into the core using 1/M approach
- Low power through ascension to power
  - Primary salt pump follows the power. Power defect, xenon, and burnup is compensated by control rods and fresh fuel addition
- Approach to equilibrium core
  - During the transition to full power, core composition will evolve: fresh fuel pebbles are added, and depleted pebbles are removed via the pebble handling and storage system (PHSS)
- All core states will operation within coolant reactivity coefficients, power per particle limits, and excess reactivity constrains

4.5 Nuclear Design: Design Basis

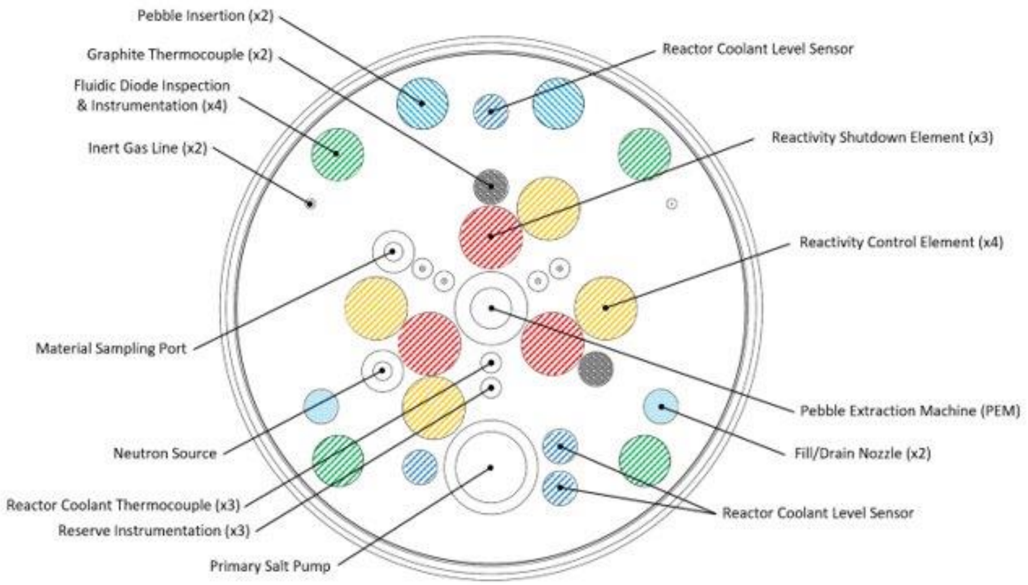
- The reactor core is designed so that the power oscillations that could result in conditions exceeding SARRDLs are not possible (PDC 12)
  - Due to the small core and the long neutron diffusion length (neutronically connected)
- The reactor core is designed so that the net effect of prompt inherent nuclear feedback tends to compensates for rapid increases in reactivity. The overall reactivity coefficient is negative. (PDC 11)
  - Large negative fuel doppler feedback
  - Positive reflector temperature coefficient due to spectrum hardening shifts flux toward core (reduces leakage) plus locally over-moderated conditions
  - Methodology used does not assume any thermal expansion of reflector (could counter-act positive feedback effect)
  - Reactivity impact due to the reflector temperature is delayed compared to fuel and coolant temperature feedback

Reactivity Coefficient	Startup	Equilibrium
Fuel Doppler (pcm/°C)	-6.2	-4.1
Moderator (pcm/°C)	-1.5	-0.4
Coolant (pcm/°C)	-2.3	-1.6
Void (pcm/%void), @3% void	-34	-53
Reflector (pcm/°C)	+2.6	+2.0

4.5 Nuclear Design: Design Basis (cont.)

- A limiting power distribution for the core design is used to ensure that the reactor core has appropriate margin to SARRDLs (PDC 10)
  - Serpent 2 used to calculate power distribution using methodology described in "KP-FHR Core Design and Analysis Methodology" (KP-TR-017-P)
  - Flux distributions are verified during startup using ex-core detectors. Flux measurements compared to predicted calculations to ensure core is operating as designed.
  - There are no consequence from control and shutdown elements not being quarter core symmetric due to the small core size and long neutron diffusion length

Power Distribution	Equilibrium
Axial Peak (F_Z)	1.2
Radial Peak (F_R)	1.2
Total Pebble Peaking (F_Q)	1.8



4.5 Nuclear Design: Design Basis (cont.)

- Shutdown elements credited to provide means to ensure SARRDLs are not exceeded, and safe shutdown is achieved; met assuming highest worth shutdown element fully withdrawn. Shutdown elements insert reactivity at a sufficient rate and amount to ensure the capability to cool the core is maintained, the reactor is shut down and can be maintained in a shutdown condition; met assuming highest worth shutdown element fully withdrawn (PDC 26, Condition 1)
- Control elements provide the capability to control reactivity changes during normal power changes, ensure SARRDLs are not exceeded and provide an independent and separate means of reactivity control from RSS. Control elements are diverse from shutdown elements (different geometry, different locations, different insertion mechanisms) (PDC 26, Condition 2)
- Shutdown elements insert reactivity at a sufficient rate and amount to ensure the capability to cool the core is maintained, the reactor is shut down and can be maintained in a shutdown condition; met assuming highest worth shutdown element fully withdrawn (PDC 26, Condition 3)
- Shutdown elements provide a means of maintaining the reactor in a shutdown state to allow for fuel loading, inspection, and repair. (PDC 26, Condition 4)

4.5 Nuclear Design: Design Basis (cont.)

- The shutdown margin calculation accounts for:
  - Power defect
  - Xenon decay
  - Operational excess reactivity
  - Margin for uncertainties

Parameter	Value at Equilibrium
Required Shutdown Margin	1,000
Actual Shutdown Margin (pcm)	3,654
Required Worth for Shutdown (pcm) [1]	11,578
Worth of Shutdown Elements (pcm)	14,232

1. Required worth considers highest worth shutdown element withdrawn (which is 6,266 pcm)

4.5 Nuclear Design: Interfaces

- The output from nuclear design is used in interfaces with other calculations
- Vessel Fluence – Supports reactor vessel design
  - Fluence on vessel accounts for core, pebble insertion and extraction lines. Fluence is attenuated by the core barrel, reflector and coolant
  - Preliminary best estimate dpa + uncertainty is within 30% of the low-level irradiation value provided in "Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor" (KP-TR-013-P)
- Nuclear Transient Analyses – Supports safety analysis
  - Conservative values used for power distribution, reactivity coefficients and shutdown margin provided as initial conditions for postulated reactivity transient events
- Core Design Limits – Supports technical specifications
  - Core design parameters during normal operation are within the fuel qualification envelope for peak fluence, peak particle power, burnup and peak fuel temperature
  - Shutdown margin
  - Coolant outlet temperature
  - Moderator pebble to fuel pebble ratio

Enclosure 2
Presentation Slides for the March 24, 2023
ACRS Kairos Power Subcommittee Meeting
(Non-Proprietary)

Hermes PSAR 4.3 Reactor Vessel System

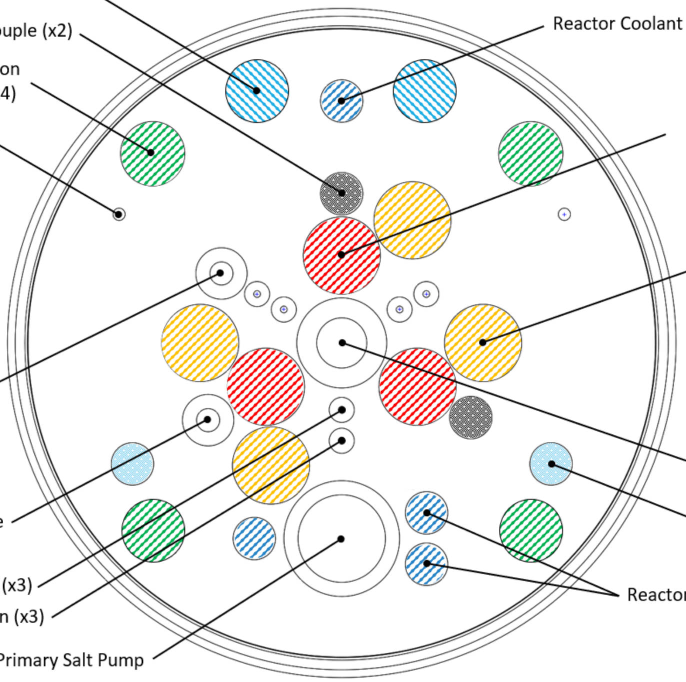
ODED DORON – SENIOR DIRECTOR, REACTOR SYSTEM DESIGN

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 24, 2023

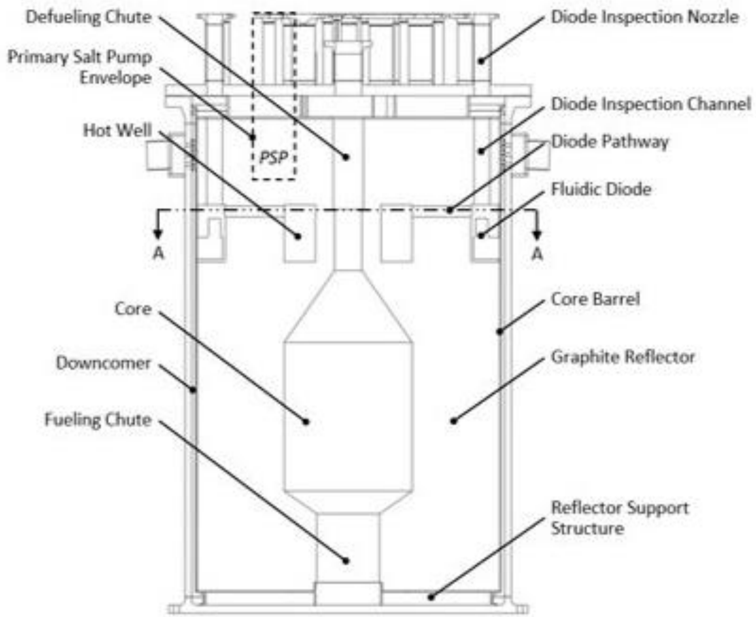
4.3 Reactor Vessel System

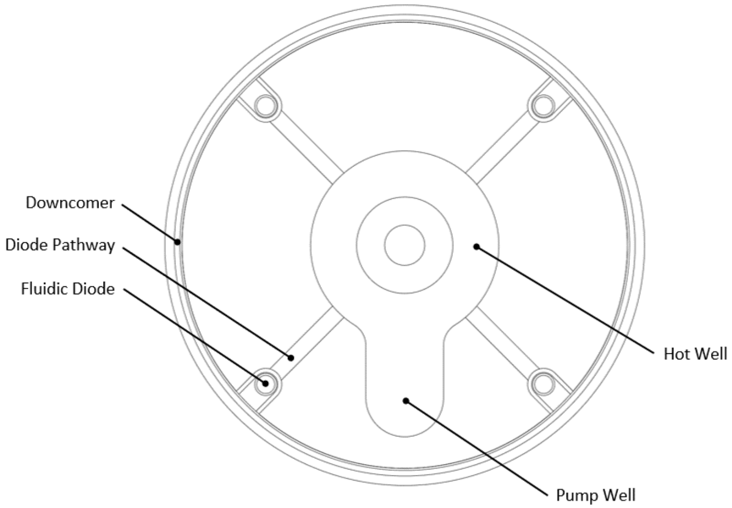
- 316H stainless steel reactor vessel bottom head, shell and top head
- Vessel material qualified per topical report "Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor" (KP-TR-013)
- Reactor Vessel Top Head
  - Supports attachment of equipment and components
  - Bolted and flanged
  - Designed to be leak tight (not credited)
  - The head, nozzles, and attachments are seismically qualified
  - 316H SS hold-down provides structural support against upward buoyant loads
- Reactor Vessel Shell
  - Maintains reactor coolant boundary
  - Provides the geometry for coolant inlet and heat transfer surface for Decay Heat Removal System (DHRS)
- Reactor Vessel Bottom Head
  - Maintains the reactor coolant boundary
  - Provides flow geometry for low pressure reactor coolant inlet to the core



4.3 Reactor Vessel System: Reactor Vessel Internals

- Core Barrel
  - 316H Stainless Steel
  - Downcomer is part of the normal and natural circulation flow pathways
- Reflector Support Structure
  - 316H Stainless Steel
  - Defines the flow path into the core
  - Supports the reflector blocks
- Reflector Blocks
  - ET-10 Graphite
  - Qualified per topical report "Graphite Material Qualification for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor" (KP-TR-014)
  - Forms the fueling chute, flow channels, core, upper plenum, hot well, pump well, defueling chute, and diode pathway
  - Part of the normal circulation flow pathway
  - Diode pathway in the reflector block and 316H SS fluidic diode within the reflector block are part of the natural circulation flow path
  - Secondary metallic hold-down structure precludes damage to the natural circulation flow path from a postulated air ingress event



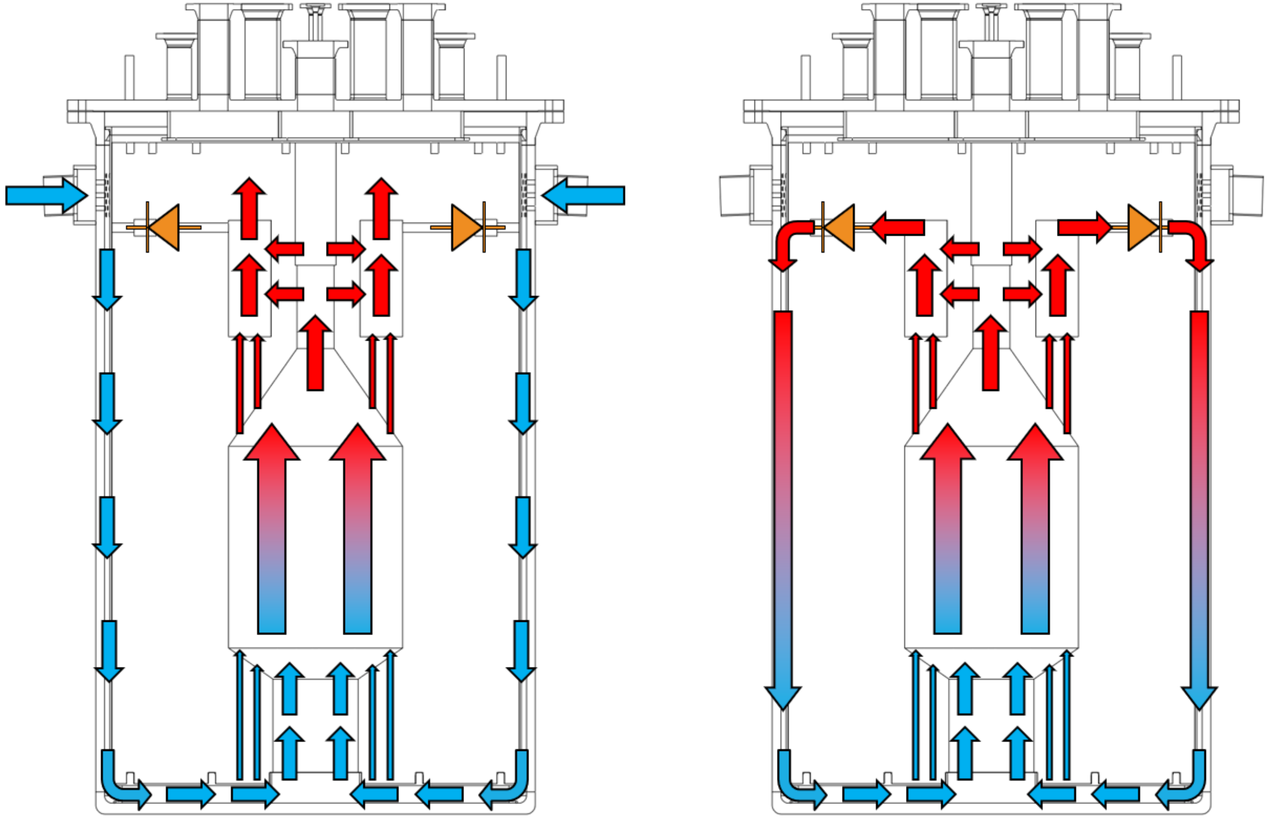


Section A-A

4.3 Reactor Vessel System: Flow Path

- Normal Circulation Flow Path (forced flow)
  - Cold leg (downcomer)
  - Reflector support structure
  - Reflector
    - Coolant inlet channels
    - Core
    - Coolant outlet channels and PEM
    - Hot well, pump well
  - Primary salt pump
  - Hot leg
  - Heat is removed by the heat rejection radiator

- Natural Circulation Flow Path (during postulated events)
  - Cold leg (downcomer)
  - Reflector support structure
  - Reflector
    - Coolant inlet channels
    - Core
    - Coolant outlet channels and PEM
    - Hot well, pump well
    - Natural circulation pathway
  - Fluidic diode
  - Cold leg (downcomer)
  - Heat is removed from the vessel wall by DHRS



(a) Normal Operation Coolant Flow Path

(b) Natural Circulation Coolant Flow Path

4.3 Reactor Vessel System: Design Basis

- Reactor vessel, reflector and 316H SS structures are designed, fabricated and tested consistent with ASME Section III, Division 5 standard (PDC 1)
- Reactor vessel, reactor vessel internals and vessel attachments are classified as SDC-3 per ASCE 43-19 to protect against failure during a design basis earthquake (PDC 2)
- Reactor vessel and vessel internals design accounts for environmental and dynamic effects like thermal expansion of vessel shell and bottom head, mechanical loadings from static weight and forces from the pebble bed, coolant and core components during start-up, normal operation and postulated events (PDC 4)
- The reflector block design maintain a geometry and coolant flow path to ensure the SARRDLs will not be exceeded by supporting coolant flow through the reflector via gaps and flow channels, thereby cooling the reflector and maintaining its structural integrity and the integrity of the coolant flow path (PDC 10)
- The reactor vessel is fabricated and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure and gross rupture and the vessel material is qualified in accordance with the metallic materials qualification topical report. The vessel is operated within as-designed operational and transient conditions and monitored for changes during in-service inspection and testing (PDC 14)
- The reactor vessel is fabricated, and tested to ASME standards, the reactor vessel design supports pre- and in-service inspections, and catch basins with sensors are used to detect leakage (PDC 30)

4.3 Reactor Vessel System: Design Basis

- The reactor vessel design has margin to withstand stresses under operating maintenance, testing, and postulated events by precluding material creep, fatigue, thermal, mechanical and hydraulic stresses that would degrade the reactor coolant boundary. Stress rupture factors encompass transient conditions and leak tight design of reactor vessel head minimizes air ingress. The design prevents fracture of the reactor coolant boundary. (PDC 31)
- The reactor vessel design permits inspection and monitoring of the structural integrity and leak-tightness of the reactor coolant boundary using the material surveillance system (MSS) to confirm irradiation assisted stress corrosion cracking is non-existent or manageable (PDC 32)
- The core barrel design maintains reactor coolant inventory in the event of a break in the primary heat transport system using anti-siphon cutouts on both sides of the core barrel. (PDC 33)
- The flow path established by the design of the reactor vessel internals support the removal of residual heat from the core to ensure SARRDLs are not exceeded during normal operation and postulated events. The physical geometry and structure of the reactor vessel internals provide a pathway for forced flow and continuous natural circulation. (PDC 34)
- The fluidic diode, reflector blocks and downcomer are designed to maintain their structural integrity in order to establish a flow path for continuous natural circulation during a postulate event. The passive cooling of the reactor core prevents damage to the vessel internals due to overheating and therefore ensures the coolable geometry of the core is maintained. (PDC 35)
- The functional capability of the natural circulation flow path is confirmed during normal operation by temperature monitoring. Appropriate periodic inspections of the fluidic diode are performed via head penetrations. (PDC 36, PDC 37)
- The reflector is qualified to maintain its structural integrity to support residual heat removal in accordance with the graphite material qualification topical report. The reactor vessel is classified as SDC-3 and will maintain its geometry to support the insertions of shutdown elements. (PDC 74)

Hermes PSAR 4.4 Biological Shield

NADER SATVAT – SENIOR MANAGER, REACTOR CORE DESIGN

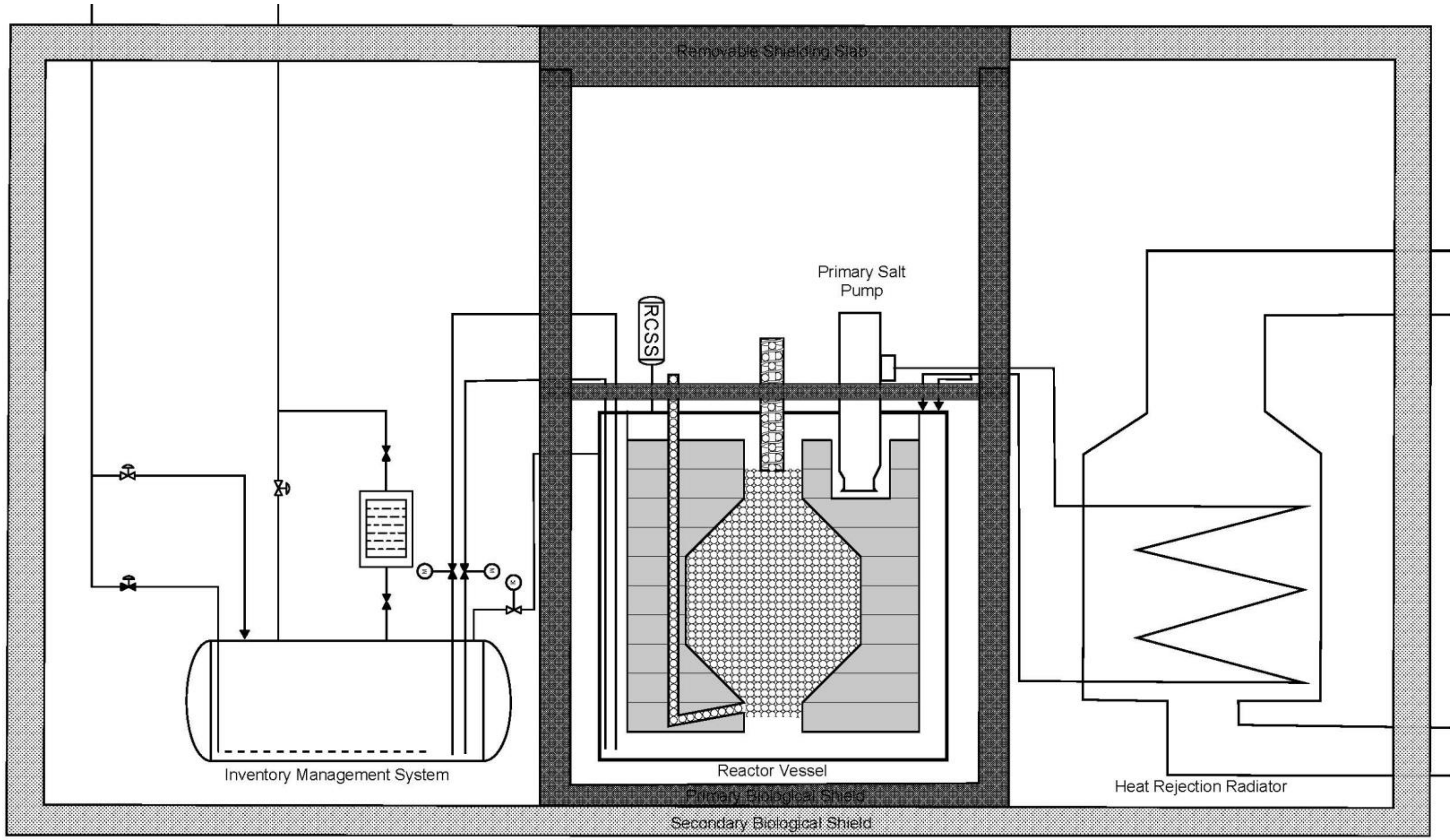
ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 24, 2023

4.4 Biological Shield

- Protects workers and the public from radiation per 10 CFR 20
- Meets radiation exposure goals in PSAR Chapter 11
- Shield Design
  - Primary shield located just outside the reactor vessel
  - Secondary shield located outside the primary shield and contains the inventory management and primary heat transfer systems
  - Both shields are concrete
- Details on biological shield will be provided as part of the operating license application

4.4 Biological Shield



Hermes PSAR 4.6 Thermal Hydraulic Design

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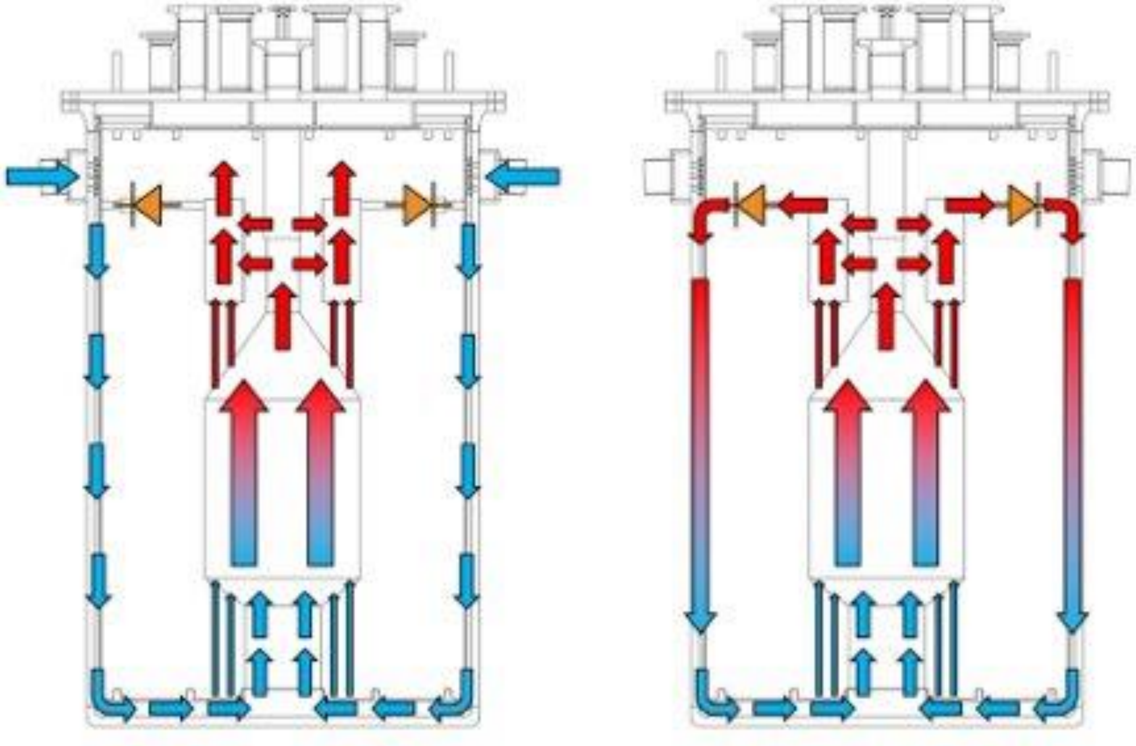
4.6 Thermal Hydraulic Design: Design Description

- The thermal hydraulic design is a combination of design features:
  - Pebble
  - Reactor coolant
  - Reactor vessel and internals
  - Primary heat transport system
- Thermal hydraulic design uses multiple heat transfer mechanisms between the reactor materials
- Thermal hydraulic design includes coolant flow path for normal operation and natural circulation
  - Natural circulation flow path uses a fluidic diode which minimizes reverse flow
  - Qualification or functional testing plans for fluidic diode and test results to validate performance will be available with the operating license
- STAR-CCM+ and KP-SAM computer codes are used in thermal hydraulic analysis of the design



(a) Normal Operation Coolant Flow Path

(b) Natural Circulation Coolant Flow Path

4.6 Thermal Hydraulic Design: Computer Codes and Models

- STAR-CCM+ is used to perform the thermal hydraulic analysis in the core design methodology
  - Steady state solver for heat transfer and fluid flow in the form of a 3-D porous media model
  - Calculates the core material temperatures used as input into the neutronics model
  - Discussed in KP-TR-017-P "KP-FHR Core Design and Analysis Methodology", Revision 1
- KP-SAM is used to perform the thermal hydraulic analysis in the postulated event methodology
  - Simplifies models to represent the major physical components and describe major physical processes (i.e., fluid flow, heat transfer)
  - Used to analyze the progression of postulated events (i.e., insertion of excess reactivity, loss of forced circulation)
  - Discussed in KP-TR-018-P "Postulated Event Methodology", Revision 2

4.6 Thermal Hydraulic Design: Design Basis

- The design provides adequate transfer of heat from the fuel to the coolant to ensure SARRDLs will not be exceeded during normal operation and postulated events (PDC 10)
- The thermal hydraulic design of the reactor system ensures that power oscillations that could result in conditions exceeding SARRDLs are not possible or can reliably and readily detected and suppressed (PDC 12)
- Residual heat is removed during normal operation and postulated events, such that SARRDLs and the design conditions of the safety-related elements of the reactor coolant boundary are not exceeded (PDC 34)
- The reactor transfers heat from the reactor core during postulated events such that fuel and reactor internal structural damage that could interfere with continued effective core cooling is prevented (PDC 35)

Hermes PSAR 4.7 Reactor Vessel Support System

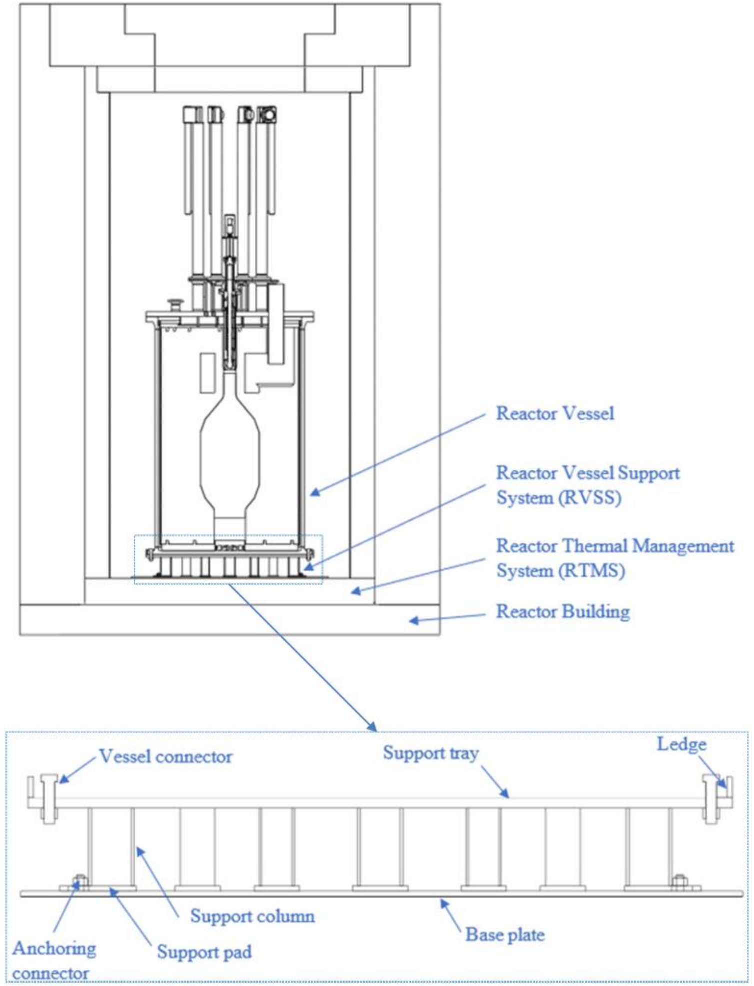
ODED DORON – SENIOR DIRECTOR, REACTOR SYSTEM DESIGN

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 24, 2023

4.7 Reactor Vessel Support System

- Reactor vessel support system (RVSS) purpose:
  - Supports the weight of the reactor vessel with fuel, coolant, internals and attachments
  - Provides thermal management to support vessel expansion
- RVSS Bottom Support:
  - Includes a support tray, ledge, support columns, support pads, base plate, vessel connector, and anchoring connector
  - 316H stainless steel
  - Reactor vessel bottom head sits directly on the bottom support
  - Designed and fabricated using ASME Section III, Division 5
  - Vertically anchored to the reactor building foundation
- RVSS Thermal Management
  - Protects the reactor building cavity concrete from thermal effects
  - Thermal break provided by insulation ensures reactor building concrete integrity
- Reactor Building Seismic Isolation
  - Does not use lateral seismic restraints for the reactor vessel and head-mounted components. RVSS designed to keep reactor vessel from uplift and shear during seismic event.
  - Design leverages seismic isolation of the reactor building to reduce seismic effects on the vessel, RVSS, and head-mounted components



4.7 Reactor Vessel Support System: Design Basis

- RVSS is designed to withstand the effects of natural phenomena and support the reactor vessel in the event of an earthquake. The bottom support and connectors meet ASCE 43-19 and preclude linear buckling in the vessel support columns and provide lateral and uplift support. (PDC 2)
- RVSS is designed for the environmental conditions including temperature loading cycles in combination with mechanical loading cycles. Catch basins with sensors for leak detectors are used to preclude damage to the RVSS from primary coolant leaks. (PDC 4)
- The RVSS design ensures the integrity of the reactor vessel during postulated events to support the geometry for passive removal of residual heat from the core by removing heat from the reactor vessel via the reactor thermal management system, actively during normal operation and passively during postulated events. (PDC 74)
- The RVSS design removes heat from the vessel and ensures the integrity of the reactor vessel and reflector blocks, thereby permitting sufficient insertion of the control and shutdown elements providing for reactor shutdown. RVSS design ensures that ACI 349-13 is met to support maintenance and inspection of the vessel bottom head and shell weld and reactor cavity. (PDC 74)

Kairos Power

Hermes PSAR Chapter 6 Engineered Safety Features

NICOLAS ZWEIBAUM – DIRECTOR, SALT SYSTEMS DESIGN

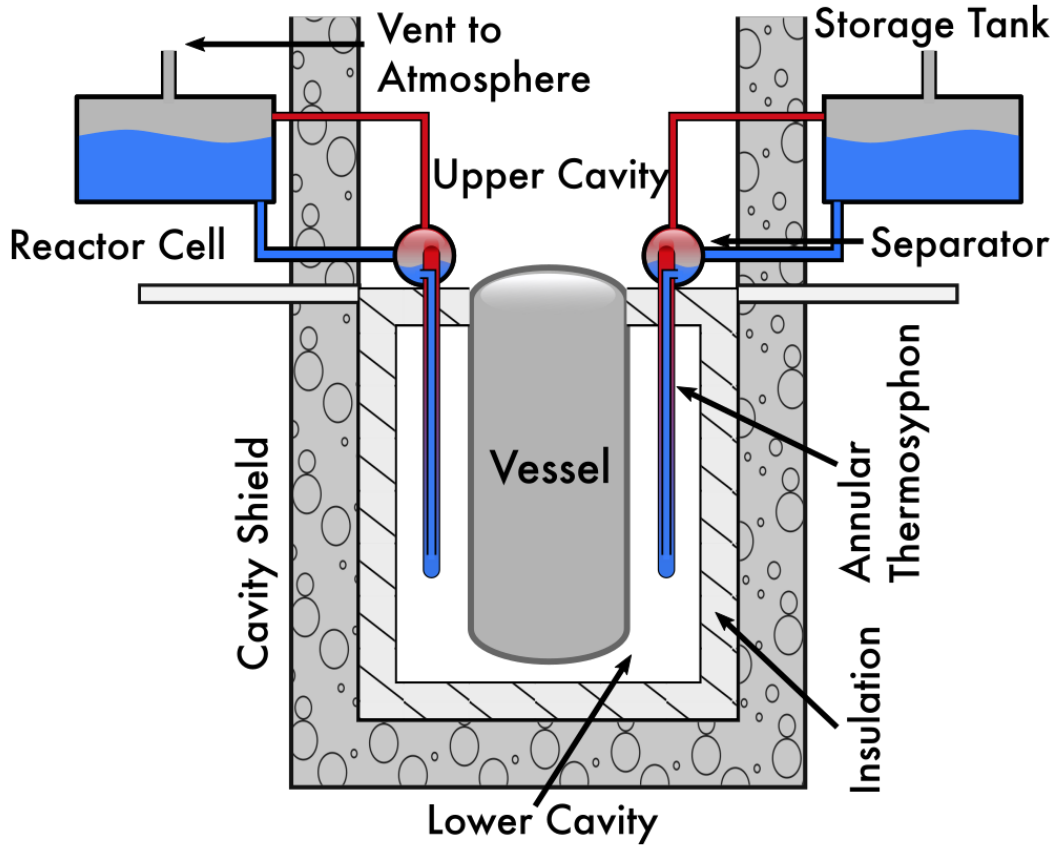
ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 24, 2023

6.2 Functional Containment

- Functional containment is defined by the NRC in SECY-18-0096 as "a barrier, or set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment"
- The functional containment for Hermes is made up of physical barriers, operating conditions, coolant design, and fuel form that limit the potential release of radioactive material
- Majority of radioactive material at risk for release is held within the design of TRISO fuel
  - Further discussion of TRISO fuel in PSAR Section 4.2
- Retention properties of Flibe act as an additional barrier for release of radionuclides for submerged fuel
  - Further discussion of the radionuclide retention capabilities of Flibe in PSAR Section 5.1
- Specified acceptable system radionuclide release design limits (SARRDLs) are met by controlling the reactor conditions (e.g., temperature and flux) that result in limiting allowable fuel conditions. Safety limits discussed in Chapter 14 will ensure SARRDLs are not exceeded, and potential dose consequences remain below dose targets.
  - SARRDLs and technical specifications will be described in the application for an operating license

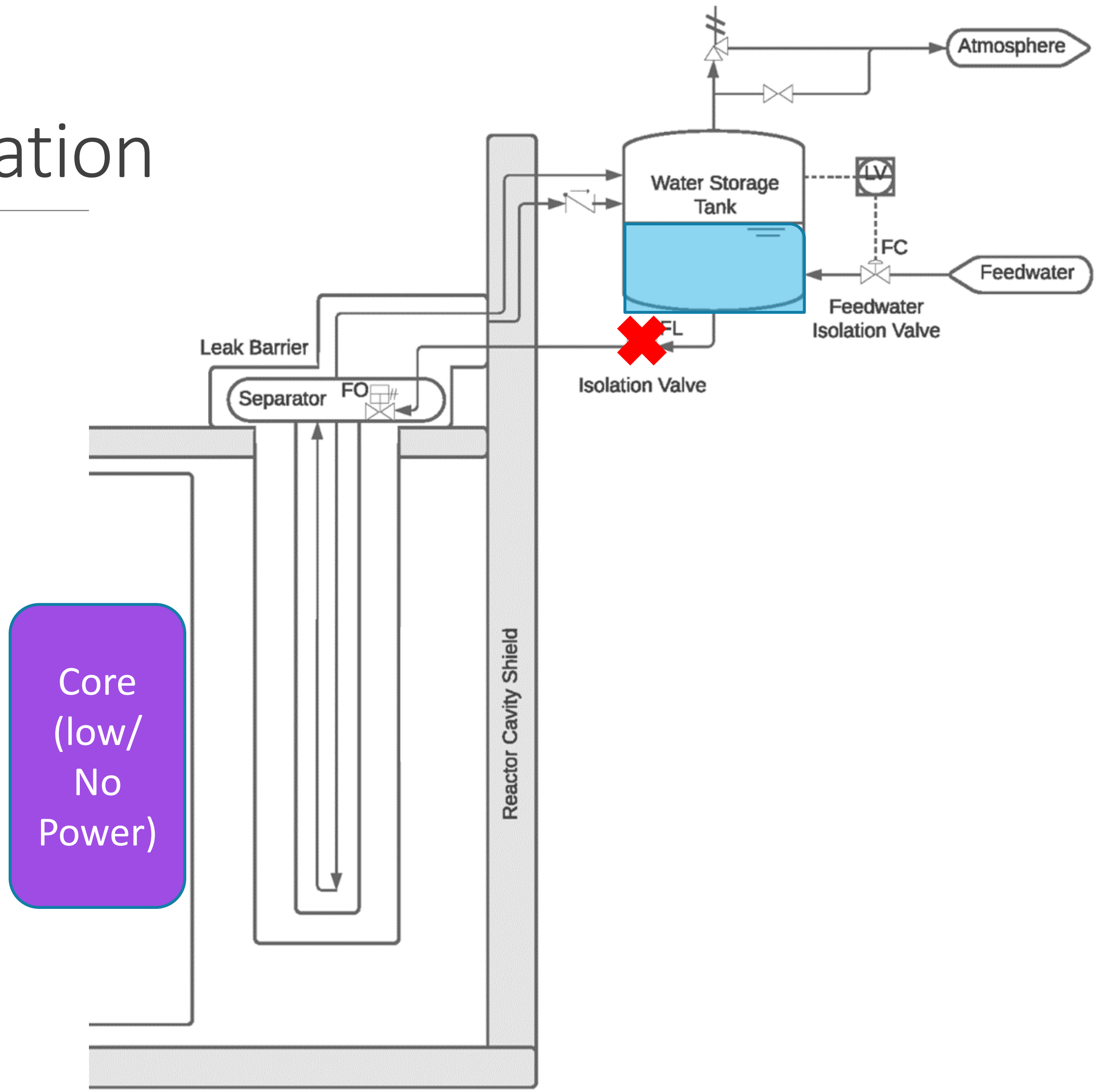
6.3 Decay Heat Removal System: Overview



- Purpose: Passive decay heat removal during postulated events when the primary heat transport system is unavailable
- Operation: DHRS is an ex-vessel system that continuously operates when the reactor is operating above a threshold power by removing energy from the vessel wall via thermal radiation and convective heat transfer to water-based annular thermosyphons (thimbles)
  - DHRS is shut off and isolated when reactor operates at low power levels (parasitic losses alone are sufficient for decay heat removal)
  - DHRS is activated when reactor starts operating above threshold power
  - No change of state when relied upon in response to postulated event
- Passive Feedback Mechanism
  - Heat removal rate is a direct function of vessel temperature due to physics of thermal radiation heat transfer

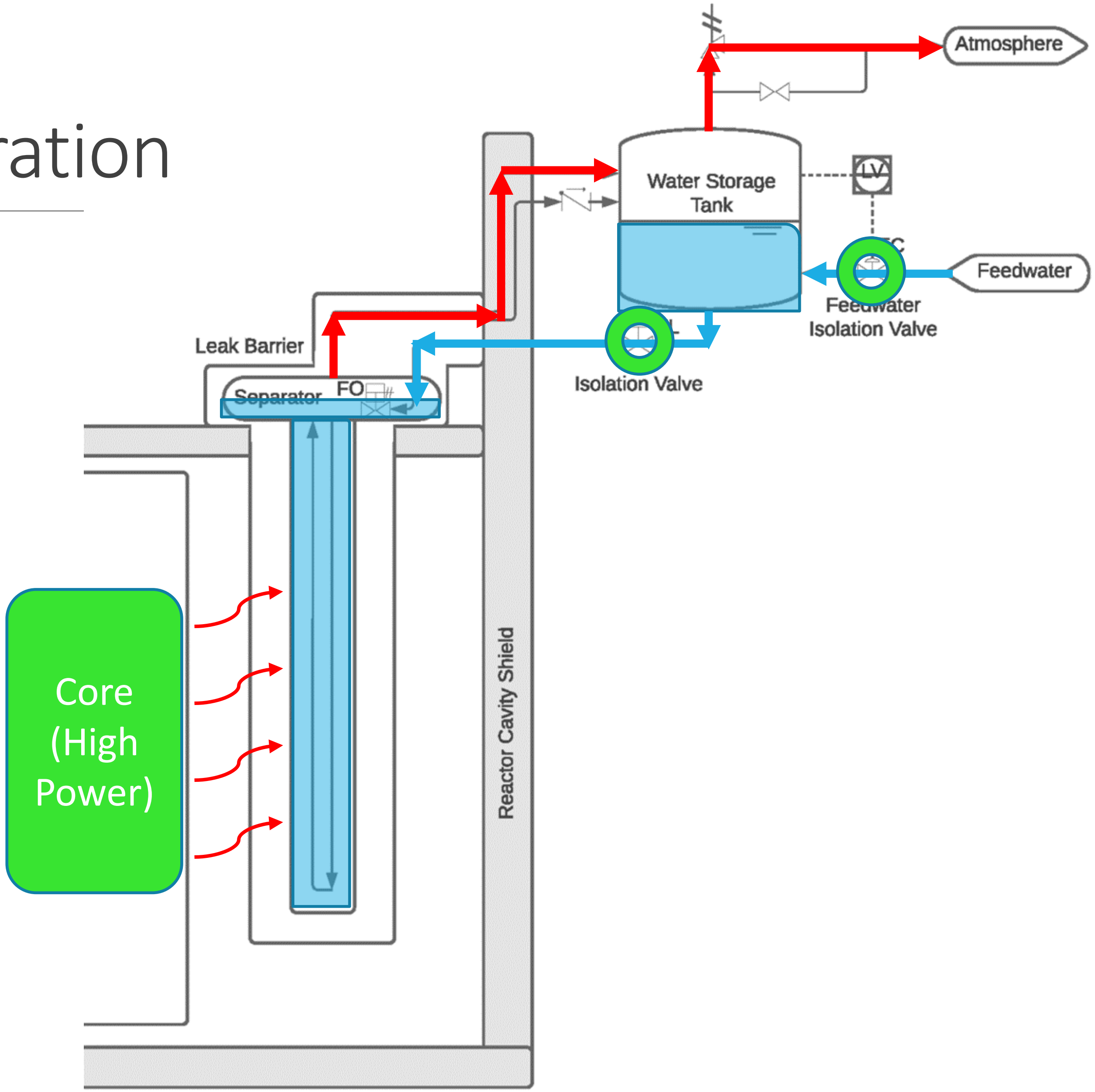
6.3 DHRS: Operation

Normal Operation
(DHRS deactivated)



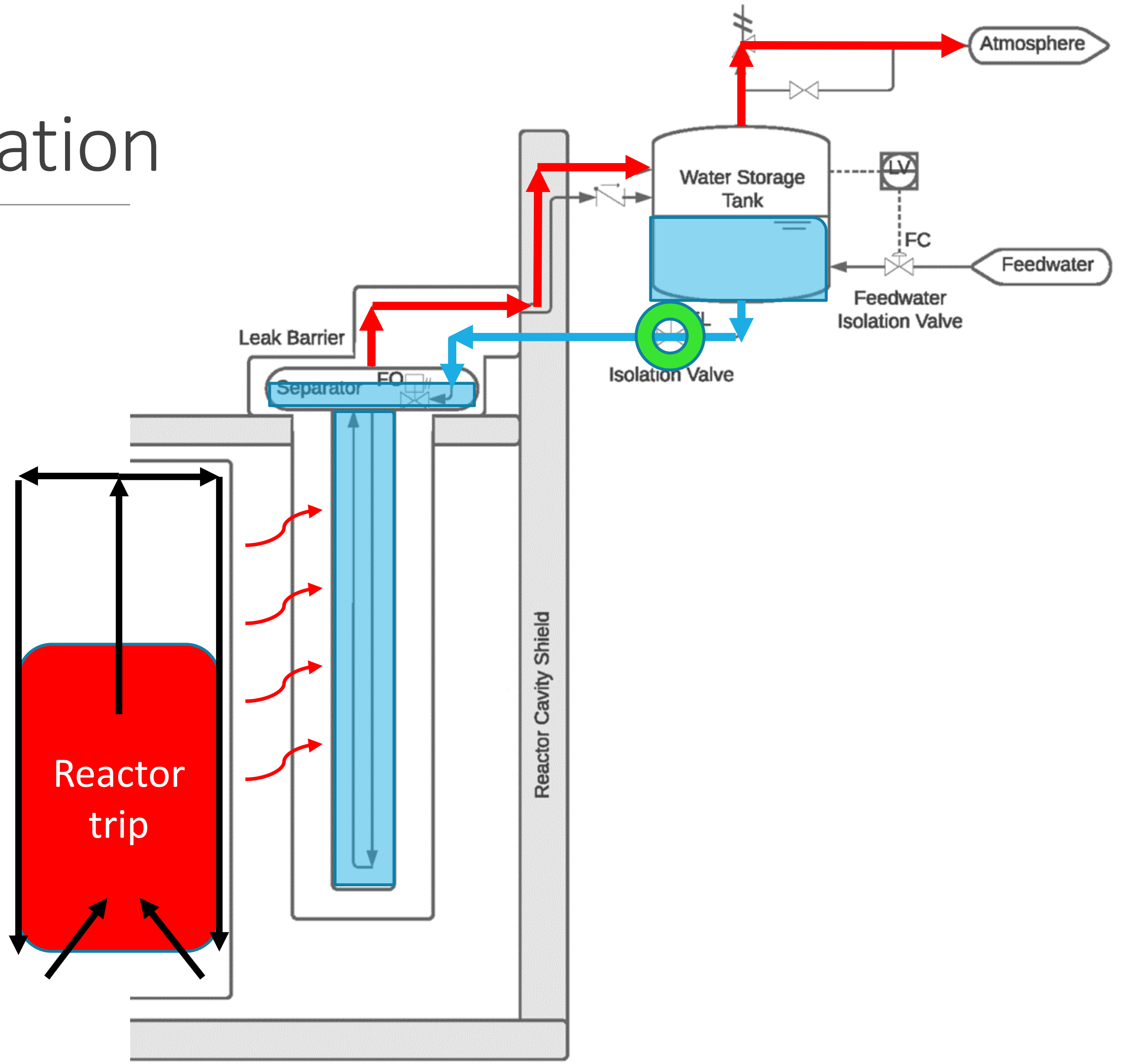
6.3 DHRS: Operation

Normal Operation

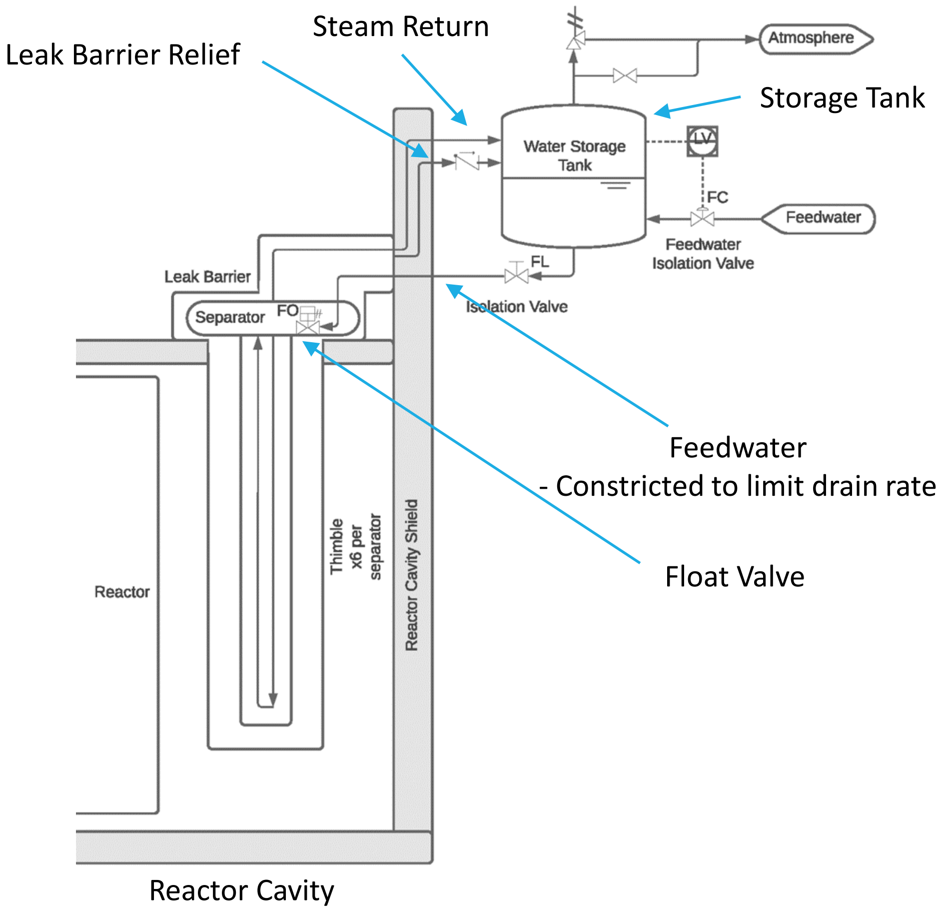


6.3 DHRS: Operation

Postulated Event
(e.g., loss of PHTS,
loss of electrical power,
loss of feedwater)



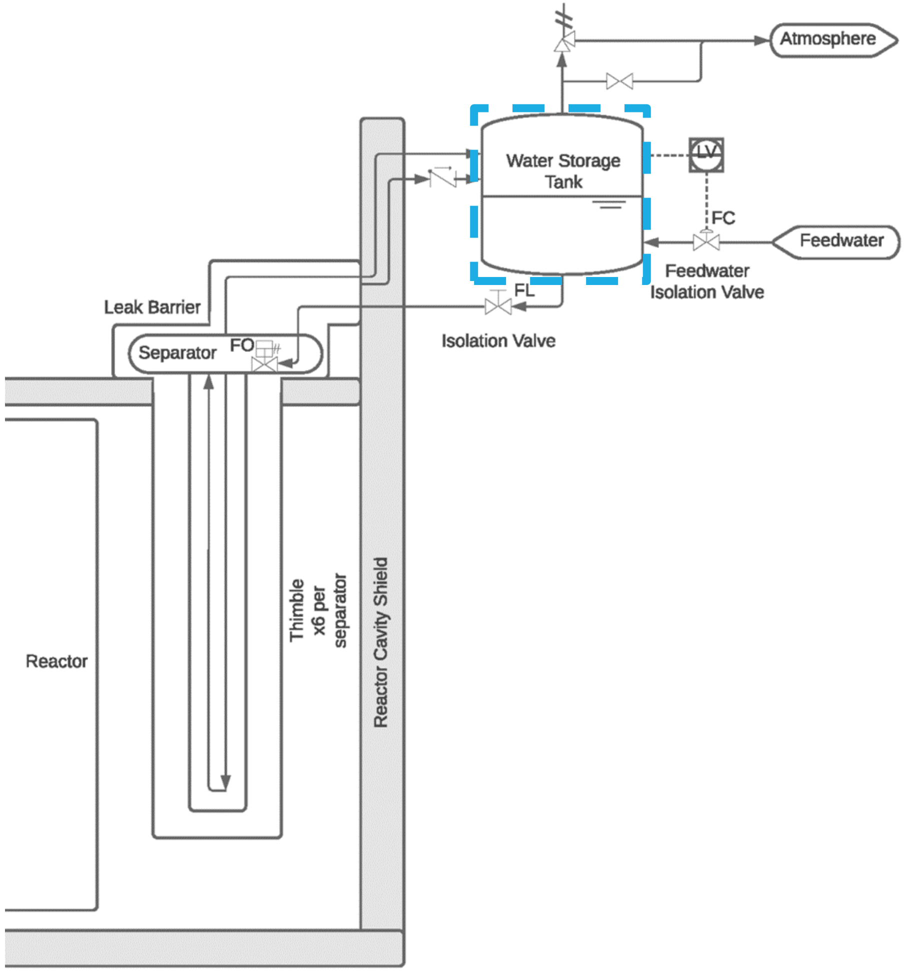
6.3 DHRS: Process Flow Diagram



- DHRS does not directly interact with reactor coolant
- No change of state on onset of postulated events
  - Always-on operation above set power levels
- Parallel and independent cooling pathways
  - Four independent cooling trains
  - Only three trains 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 fails open

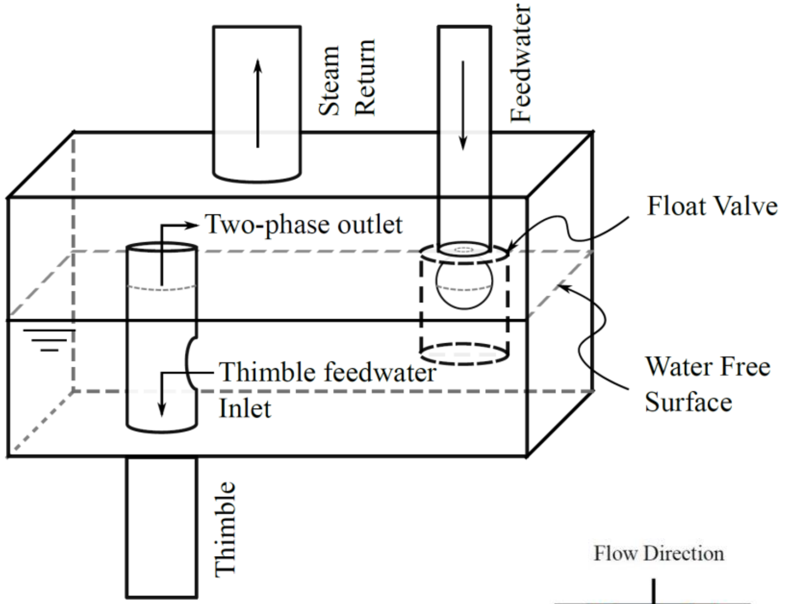
6.3 DHRS: Water Storage Tank

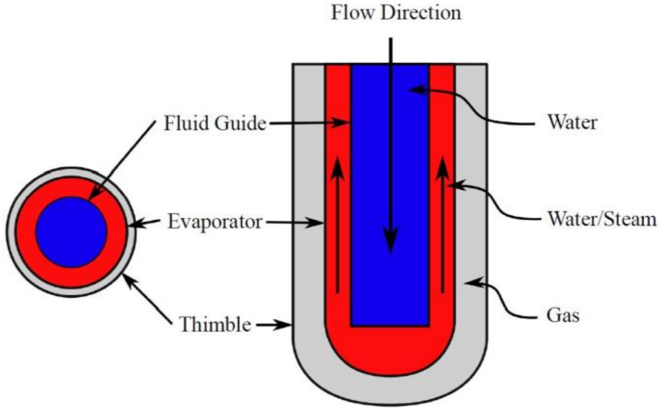
- Sizing
  - Sufficient inventory for up to 7 days of continuous operation to support heat removal to mitigate postulated events
- Location
  - Outside of reactor cavity
  - Higher elevation than other DHRS components
  - Gravity-driven flow of water to separator and thimbles
- Redundancy / Independence
  - 3 out of 4 tanks needed for adequate heat removal
  - Each tank is independent in its location and connection to thimbles



6.3 DHRS: Separator and Thimble

- Separator
  - Interface between water storage tank and thimbles
  - Float valve
    - When water level exceeds threshold value, the float valve blocks the feedwater line
    - When water level is below threshold value, the float valve allows for continuous flow
  - Passive operation and fail-open design
    - Floods separator
    - Does not affect the net heat removal performance of the thimbles
- Thimble
  - Annular thermosyphons located circumferentially around the outside of the reactor vessel
  - Guide tube located within evaporator tube
  - Leak barrier
    - Dual wall design – still can remove heat
  - Passive flow





6.3 Decay Heat Removal System: Design Basis

- Safety-related portions of DHRS are designed to ASME III Div. 5 Class B, ASCE 43-19, ASCE 4-16, and ACI 349-13 codes and standards (PDC 1)
- DHRS is primarily located in the safety-related portion of the reactor building, which is designed to protect safety-related components from external hazards. Failure of non-safety related DHRS components does not affect the performance of safety-related SSCs (PDC 2)
- DHRS is designed with low combustible materials and uses physical separation of the trains to minimize the probability and effect of fires and explosions (PDC 3)
- DHRS is designed with materials that will withstand the environmental conditions in the reactor cavity during normal operation, maintenance, testing, and postulated events. DHRS components are designed to preclude cascading failures and failures that could impact nearby safety systems (PDC 4)
- DHRS is capable of removing an adequate amount of decay heat to ensure that SARRDLs are not exceeded, and reactor vessel and fuel temperatures remain below their design limits (PDC 34 and 35)
- DHRS is designed to allow for periodic inspection and functional testing to ensure integrity, operability, and performance of the system (PDC 36 and 37)

Hermes PSAR Chapter 9 Auxiliary Systems

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 24, 2023

NICO ZWEIBAUM – DIRECTOR, SALT SYSTEMS DESIGN
KIERAN DOLAN – SENIOR ENGINEER, FUELS AND MATERIALS
ANDREW LINGENFELTER – SENIOR ENGINEER, ENGINEERING DESIGN
ANTHONIE CILLIERS – DIRECTOR, INSTRUMENTATION AND CONTROLS
MARGARET ELLENSON – MANAGER, NUCLEAR FACILITIES AND SAFEGUARDS AND SECURITY

9.1 Reactor Auxiliary Systems Overview

- The reactor coolant auxiliary systems are a collection of systems that provide support for the functionality and performance of Flibe:
  - Remove fission products, activation products, and other chemical impurities and particulates from the reactor coolant
  - Maintain the cover gas atmosphere (pressure and composition) in the head space above the core
  - Provide removal and storage of tritium
  - Control inventory, filling, and draining processes for systems containing reactor coolant, including transfer of coolant into the reactor
  - Provide active and passive thermal management to reactor system components
- These functions are implemented into the following reactor coolant auxiliary systems:
  - Chemistry control system
  - Inert gas system
  - Tritium management system
  - Inventory management system
  - Reactor thermal management system
- These systems are not credited with performing any safety-related functions

9.1.1 Chemistry Control System

- The CCS is not credited with performing any safety-related functions
- The CCS monitors primary coolant chemistry for compliance with Flibe specifications
- The CCS extracts coolant samples for an offline analysis of the Flibe chemistry
- Due to the proximity of the CCS to the reactor vessel, the CCS is designed so that seismic induced failure does not impact the reactor vessel system (PDC 2)
- Due to the proximity of the CCS to the reactor vessel, the CCS is designed so that adverse effects of postulated CCS failures do not impact the reactor vessel system (PDC 4)
- The CCS monitors the reactor coolant purity with offline sampling analysis to determine if the reactor coolant is within specified design limits (PDC 70)
- Consistent with 10 CFR 20.1406, the CCS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning

9.1.2 Inert Gas System

- The IGS is not credited with performing any safety-related functions
- Provides inert argon gas as a purging flow to system components during normal operation and maintenance
- Removes impurities from the cover gas
- Provides reactor coolant motive pressure during filling and draining operations
- The IGS may be in proximity or connected to safety-related SSCs and may cross the seismic isolation moat. The IGS is designed so that seismic induced failure does not impact safety-related SSCs from performing their safety function (PDC 2)
- The IGS is a low-pressure system and precludes pipe whip. Nearby safety-related SSCs will not be affected by escaping inert argon gas (PDC 4)
- The IGS monitors radioactivity levels in the gas to support the evaluation of the radioactive material releases that might occur as a result of a system or fuel failure (PDC 64)
- Consistent with 10 CFR 20.1406, the IGS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning

9.1.3 Tritium Management System

- The TMS is not credited with performing any safety-related functions
- TMS separates tritium from argon in the inert gas system and from dry air in the Reactor Building cells
- TMS collects and temporarily stores tritium for final disposition
  - In accordance with 10 CFR 71.51, Type A and Type B packaging canisters are used
- Due to the potential proximity of the TMS to the reactor vessel, the TMS is designed so that seismic-induced failure does not impact the reactor vessel system (PDC 2)
- Tritium monitoring sensors are selected to provide measurements over a range of anticipated tritium activities where measurements are needed (PDC 13)
- The TMS maintains a minimum level of overall tritium capture capacity in order to minimize tritium releases from the plant (PDC 60)
- Radiation monitoring is provided in the TMS for the evaluation of tritium levels in TMS subsystems in support of evaluation of radioactive material releases that might occur as a result of a system failure (PDC 64)
- Consistent with 10 CFR 20.1406, the TMS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning

9.1.4 Inventory Management System

- The IMS is not credited with performing any safety-related functions
- The IMS maintains primary coolant level in the reactor vessel during normal operations
- The IMS fills and drains the reactor vessel and the primary heat transport system during start-up and shutdown operations
- The IMS maintains primary coolant purity by replacing circulating salt with new salt
- Due to the proximity of the IMS to the reactor vessel, the IMS is designed so that seismic-induced failure does not impact the reactor vessel system (PDC 2)
- Due to the proximity of the IMS to the reactor vessel, the IMS is designed so that adverse effects of postulated IMS failures do not impact the reactor vessel system (PDC 4)
- The IMS includes design features to limit the loss of reactor vessel coolant inventory in the event of breaks in the system (PDC 33)
- The IMS may be used to remove and replace a sufficient amount of reactor coolant to restore conformance to the Flibe specification (PDC 70)
- Consistent with 10 CFR 20.1406, the IMS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning

9.1.5 Reactor Thermal Management System

- The RTMS is not credited with performing any safety-related functions
- Nearby safety-related SSCs are protected from RTMS failure in the event of an earthquake (PDC 2)
- The RTMS uses water cooling to transfer heat from SSCs to the ultimate heat sink during normal operation and maintains the operational temperature limits of concrete structures during normal operations (PDC 44)
- The system is designed to permit periodic appropriate inspections and testing to ensure integrity and capability to cool SSCs and to ensure adequate interface with other systems supporting heat transfer to the ultimate heat sink (PDC 45, PDC 46)
- RTMS is designed to pre-heat the reactor vessel and to ensure Flibe in the vessel is maintained above a minimum operating temperature (PDC 71)
- Consistent with 10 CFR 20.1406, the RTMS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning

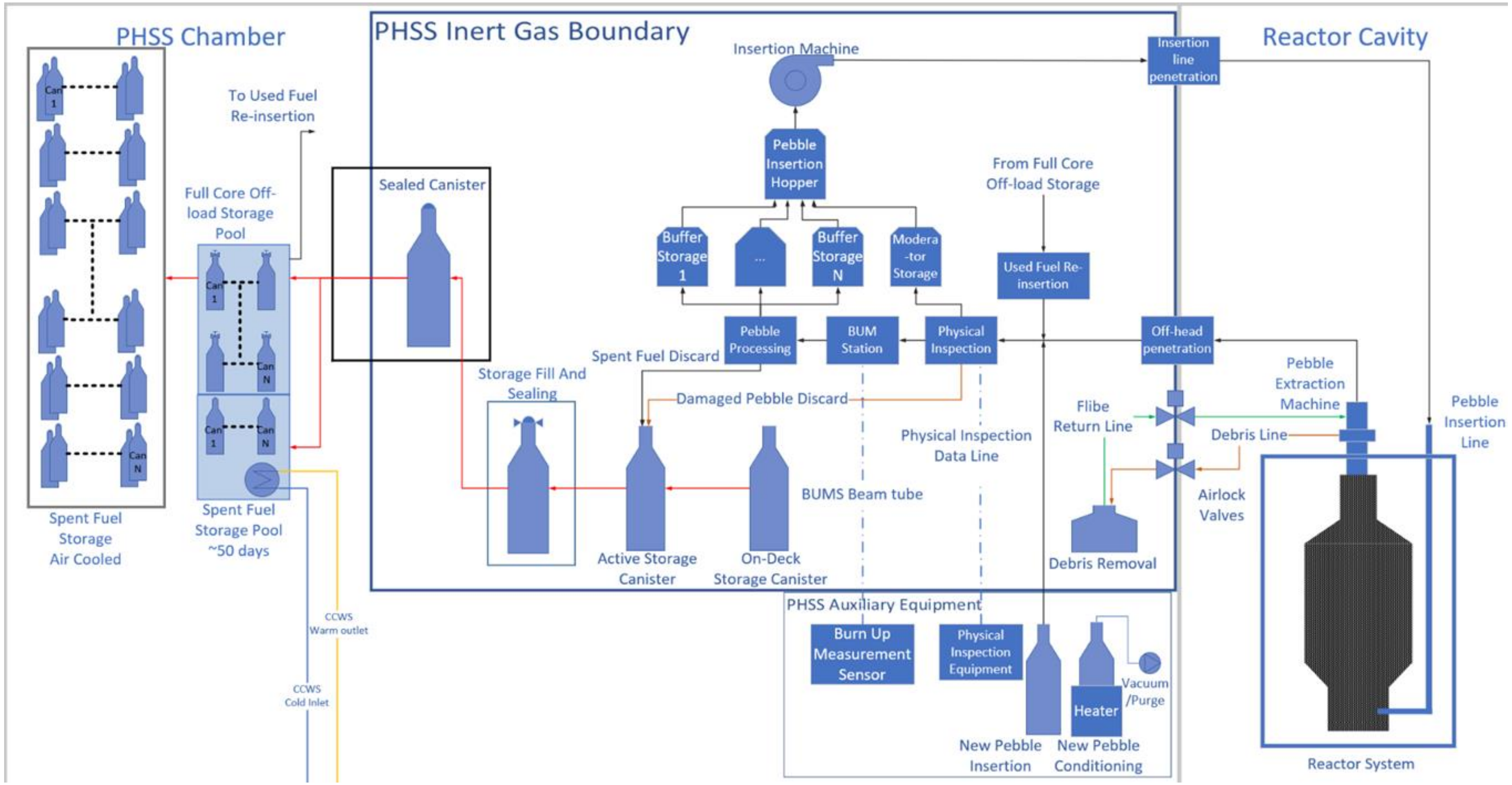
9.2 Reactor Building Heating, Ventilation, and Air Conditioning System

- The RBHVAC is not credited to perform any safety-related functions
- Reactor building heating, ventilation, and air conditioning system provides independent environmental control to the reactor building
  - The system is designed to ensure occupational dose does not exceed 10 CFR 20 limits
- Consistent with 10 CFR 20.1406, the RBHVAC is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning
- RBHVAC does not adversely affect safety-related SSCs located nearby (PDC 2)
- RBHVAC is designed to control the release of radioactive materials in gaseous effluents during normal operation (PDC 60)
- RBHVAC is designed to provide for monitoring of the RB effluent discharge paths for radioactivity that may be release during normal operation (PDC 64)

9.3 Pebble Handling and Storage System

- 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 mechanism removes pebbles from molten salt
  - Pebble Inspection: Performs flaw detection and burn-up measurement of removed pebbles
  - Processing: Sorts pebbles into appropriate buffer storage channel based on pebble type
  - Insertion: Stepper wheel feeder mechanism inserts pebbles into the reactor via an in-vessel insertion line
  - Storage Canister: Stores ~2,000 fuel pebbles in a non-critical configuration
  - Storage Cooling Area: In-building storage area for spent fuel canisters, capable of passive cooling during loss of power and other postulated events
  - New Pebble Introduction: Stores fresh fuel and prepares fuel for circulation via a high-temperature bakeout

9.3 Pebble Handling and Storage System



9.3 Pebble Handling and Storage System: Design Basis

- Storage bay, pool and support restraint structures in the pool are designed as seismic category SDC-3 to ensure geometry is maintained in the event of an earthquake (PDC 2)
- System limits grinding of pebbles and the accumulation of graphite dust to minimize the potential of fire and explosion (PDC 3)
- The canister design considers environmental conditions such as pressure accumulation of radionuclides and thermal loads; the canister interior accounts for radiolysis products. The system design accounts for complete submergence and internal flooding of the storage canisters in water. (PDC 4)
- An anti-siphon feature on the pebble insertion line limits inventory loss to primary salt pump elevation, the PEM is above the coolant free surface (PDC 33)
- The TRISO particle confines radioactive material rather than the PHSS and pebble loads do not introduce incremental particle failures thereby ensuring the PHSS does not act to confine or contain radioactivity (PDC 61)
- The design prevents criticality by controlling pebble removal rate. The system design precludes moisture intrusion and handling equipment maintains geometry via interlocks. (PDC 62)
- The inspection and sorting function ensures damaged pebbles are removed from use (PDC 63)

9.4 Fire Protection Systems and Programs

- The fire protection system is not credited with performing any safety-related functions
- Designed to detect and extinguish fires so that a continuing fire will not prevent safe shutdown (PDC 3)
- Noncombustible and fire-resistant materials are used whenever practical, particularly in locations with SSCs that are safety-related or required for safe shutdown (PDC 3)
- The fire protection system will conform to local building and fire codes, ANSI/ANS 15.17 “Fire Protection Program for Research Reactors,” NFPA 801, and Life Safety Code NFPA 101
- The system is designed so that seismic induced failure does not impact nearby safety-related SSCs (PDC 2)

9.5 Communication

- The communication system is not credited with performing any safety-related functions
- Composed of diverse and independent subsystems:
  - Plant radio
  - Public address and general alarm
  - Communication capability in the event of a loss of normal power
  - Distributed antenna
  - Security communication
  - In addition, diverse commercial communication systems will be utilized for on- and off-site communication
- Used for normal and emergency conditions to communicate between key areas of the facility
- Phone lines area available for off-site communication in the case of an emergency

9.6 Possession and Use of Byproduct, Source, and Special Nuclear Material

- Byproduct material is managed by compliance with 10 CFR Part 30 , by use of spent fuel canisters, by the tritium management system, and by the radioactive waste management program
- Source material is managed by compliance with 10 CFR Part 40, by use of fresh and spent fuel canisters, and by the nature of the pebble design, in which the source material is encapsulated in a graphite substrate
- Special nuclear material (SNM) is managed by compliance with 10 CFR Part 70, by the use of fresh and spent fuel canisters, by the pebble handling and storage system, which includes shielding, by the reactor vessel, and by the nature of the pebble design, in which the SNM is encapsulated in a graphite substrate
- Requests for materials licenses will be submitted at a future date

9.7 Plant Water Systems

- The water systems are not credited with performing any safety-related functions
- Service water system is the facilities main supply of water and provides water to the treated water system
- Treated Water System provides chemistry control of the service water and provides water to the component cooling water, chilled water, and decay heat removal systems
  - Treated water is designed to protect against design basis earthquake for nearby safety-related SSCs (PDC 2)
  - Nearby safety-related SSCs are protected from the effects of discharging fluid and missiles and precluded from pipe whip hazards by design (PDC 4)
- Component cooling water system provides water cooling for reactor building HVAC systems, the equipment and structural cooling system, spent fuel cooling system, and the inert gas system coolers and compressors
  - The system is designed to protect against design basis earthquake for nearby safety-related SSCs (PDC 2)
  - Nearby safety-related SSCs are protected from the effects of discharging fluid and missiles and precluded from pipe whip hazards by design (PDC 4)
  - The system is designed with the capability to isolate leaks, permit appropriate periodic inspection and testing to ensure the integrity and capability of the system to cool SSCs, and to adequately transfer heat to the ultimate heat sink (PDC 44, 45, and 46)
- Chilled water system provides cooling water for nonessential heat loads
- Consistent with 10 CFR 20.1406, the plant water systems that directly interface with the systems that contain radioactive material are designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning

9.8.1 Remote Maintenance and Inspection System

- The remote maintenance and inspection system (RMIS) is not credited with performing any safety-related functions
- The RMIS can remotely handle components in the reactor system, PHTS, and PHSS
- RMIS supports the following maintenance activities:
  - Disassemble flanges and subassemblies
  - Remove subassemblies
  - Clear fuel and residual coolant before removal of SSCs for maintenance
  - Transport of equipment to hot maintenance cells (via use of shielded casks)
  - Activities performed in standalone hot cells
  - Use of through-wall electro-mechanical manipulators for hot cells
  - Use of cranes for hot cell and post-irradiation examination facilities
- Consistent with 10 CFR 20.1406, the RMIS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning
- The capabilities of RMIS will limit the personnel occupational exposures to below 10 CFR Part 20 limits

9.8.2 Spent Fuel Cooling System

- The spent fuel cooling system (SFCS) is not credited with performing any safety-related functions
- The SFCS provides forced air cooling for spent fuel storage canisters in the storage bay of the PHSS and recirculates water in the spent fuel pool
  - Consists of fans and piping that remove heat during normal operation
  - Maintains desired operational temperatures in the storage bay
- In the event normal power is not available, the SFCS is capable of passively cooling spent fuel storage canisters
- The system is designed to ensure nearby safety-related SSCs are protected from seismic-induced failure (PDC 2)
- Nearby safety-related SSCs are protected from dynamic effects such as missiles by design (PDC 4)
- Consistent with 10 CFR 20.1406, the SFCS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning

9.8.3 Compressed Air System

- The compressed air system is not credited with performing any safety-related functions
- The compressed air system provides and distributes compressed air for maintenance and use in valve operation
- The system is designed so that a failure of the system does not interfere or preclude the ability of a safety-related system to perform its safety function
- The system does not directly interface with systems that contain or have the potential to contain radioactive materials

9.8.4 Cranes and Rigging

- The crane and rigging is not credited with performing any safety-related functions
- A crane and rigging are provided to lift and move equipment within the reactor building, facilitate receiving and shipping, and support maintenance activities
- The system is designed to ensure nearby safety-related SSCs are protected from seismic induced failure (PDC 2)
- Nearby safety-related SSCs are protected from dynamic effects by design, such as administrative controls and interlocks (PDC 4)
- Implements codes and standards from ASME B30.2-2016

9.8.5 Auxiliary Site Services

- Auxiliary site services is not credited with performing any safety-related functions
- The following services provide additional functions necessary to maintain and operate the plant:
  - Machine shop(s), which include radioactive and non-radioactive machining capabilities
  - Chemistry laboratory
  - Post-irradiation examination laboratory
  - Materials testing laboratory
  - Vents and drains for non-potentially contaminated facility compartments
  - Warehouse(s) for storage of spare equipment
  - Storage of contaminated equipment
  - Facility lighting, including emergency lighting
  - Non-hazardous waste management services
  - Firewater storage systems
  - Storm and sanitary sewers
  - Groundwater monitoring wells
- The system is designed to ensure nearby safety-related SSCs are protected from seismic induced failure (PDC 2)
- The capabilities of the Auxiliary Site Services will limit the personnel occupational exposures to below 10 CFR Part 20 limits
- Services that involve handling of radioactive material may include remote manipulation capabilities, as appropriate, to facilitate limiting personnel occupational exposures to below 10 CFR Part 20 limits

NRC Staff Review for PSAR Chapter 1
The Facility

Briefing for the Advisory Committee on Reactor Safeguards

Thursday, March 23, 2023

By the Division of Advanced Reactors and Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Agenda

- Introduction
- Regulatory Requirements
- CP Application Review Overview
- Summary of Chapter 1 Review

Introduction

- Kairos Power LLC has requested a construction permit for a 35 MWth* non-power reactor facility known as Hermes
  - **Purpose**: test and demonstrate key technologies, design features, and safety functions for the commercial Kairos Power fluoride salt-cooled, high temperature reactor (KP-FHR) structures, systems, and components (SSCs)
  - **Location**: East Tennessee Technology Park near Oak Ridge
- Hermes would be licensed as a non-power reactor under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50
  - Class 104 license per 10 CFR 50.21(c) for testing research and development

*MWth = megawatts of thermal power

Regulatory Requirements

- The staff's evaluation of Hermes' preliminary design and analysis was based primarily upon the following 10 CFR requirements:
  - 10 CFR 50.2, "Definitions."
  - 10 CFR 50.21, "Class 104 licenses; for medical therapy and research and development facilities," paragraph (c).
  - 10 CFR 50.33, "Contents of applications; general information," paragraph (f).
  - 10 CFR 50.34, "Contents of applications; technical information," paragraph (a), "Preliminary safety analysis report."
  - 10 CFR 50.35, "Issuance of construction permits."
  - 10 CFR 50.40, "Common standards."
  - 10 CFR 50.41, "Additional standards for class 104 licenses."
  - 10 CFR 50.50, "Issuance of licenses and construction permits."
  - 10 CFR 50.55, "Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses."
  - 10 CFR 50.58, "Hearings and report of the Advisory Committee on Reactor Safeguards."
  - 10 CFR Part 50, Appendix C, "A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Construction Permits and Combined Licenses."
  - 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

CP Application Review Overview

- Preliminary Safety Analysis Report, Revision 2, dated February 24, 2023 (ADAMS Accession No. ML23055A672)
- Construction Permit
  - Allows licensee to proceed with construction based on preliminary design information
  - Does not approve of the safety of any design feature or specification unless specifically requested by the applicant
- Regulatory Guidance and Acceptance Criteria
  - NUREG 1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors”
  - Other guidance (e.g., regulatory guides and ANSI/ANS standards) and engineering judgment used, as appropriate, to make construction permit findings

Summary of Chapter 1 Review

- PSAR Section 1.1, “Introduction”
- PSAR Section 1.2, “Summary and Conclusions on Principal Safety Considerations”
  - Applicable standards and requirements of the Atomic Energy Act and NRC regulations have been met.
- PSAR Section 1.3, “General Description”
- PSAR Section 1.4, “Shared Facilities and Equipment”
  - No existing facilities or equipment will be shared by Hermes, and any site infrastructure that may be shared is not needed to perform a safety function.

Summary of Chapter 1 Review

- PSAR Section 1.6, "Summary of Operations"
  - Kairos' preliminary information on proposed Hermes operation is consistent with relevant assumptions and analyses later in the PSAR in which any safety implications of the proposed operations are evaluated.
- PSAR Section 1.7, "Compliance with the Nuclear Waste Policy Act of 1982"
  - Kairos is in compliance with the NWPA for the CP stage based on its documentation of communication with the Department of Energy to enter into a contract for the disposition of high-level waste.
- PSAR Section 1.8, "Facility Modifications and History"
  - There are no existing facilities or modifications with respect to the Hermes facility.

Ongoing Research and Development (PSAR Section 1.3.9)

- 10 CFR 50.34(a)(8) requires identification of SSCs requiring R&D
- PSAR Section 1.3.9 identifies the following R&D activities:
  - Confirm fuel pebble behavior (PSAR Section 4.2.1).
  - Material surveillance sampling program for the reactor vessel and internals (PSAR Section 4.3.4).
  - Testing of high temperature material to qualify Alloy 316H and ER16-8-2 (PSAR Section 4.3).
  - Analysis of potential graphite oxidation in certain postulated (PSAR Section 4.3).
  - Validation of computer codes for core design and analysis methodology (PSAR Section 4.5).
  - Development and qualification testing of a fluidic diode device (PSAR Section 4.6).
  - Justification of thermodynamic data and associated vapor pressure correlations of representative species (PSAR Section 5.1.3).
  - Develop process sensor technology for key reactor process variables (PSAR Section 7.5.3).
  - Develop the reactor coolant chemical monitoring instrumentation (PSAR Section 9.1.1).
- Activities will be completed before the completion of construction, which is expected by December 2026
- The staff is tracking these activities and will verify their resolution prior to the completion of construction

PSAR Section 1.5, “Comparison with Similar Facilities”

- Key Similar Facilities
  - Molten Salt Reactor Experiment (MSRE) – molten fluoride salt coolant (with liquid fuel)
  - Pebble Bed Reactors (PBRs) - use of pebbles with TRISO fuel particles.
  - High temperature gas reactors (HTGRs) at Peach Bottom 1 and Ft. St. Vrain - TRISO particle fuel in non-pebble form
  - Advanced Gas Reactor (AGR) in the United Kingdom - use of graphite as a neutron moderator

- The staff finds:
  - Kairos has compared the design bases and safety considerations of Hermes with similar facilities.
  - Aspects of the Hermes design that are similar to features in other facilities should be expected to perform in a similar manner to these comparable features in other facilities.
  - Kairos is using test data and operational experience from facilities with similar components and design features in designing Hermes components, as practicable.

Questions?

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

NRC Staff Review for PSAR Sections 3.1 and 3.6

Briefing for the Advisory Committee on Reactor Safeguards

March 23, 2023

By the Division of Advanced Reactors and Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Agenda

- PSAR Section 3.1, “Design Criteria”
  - Overview and Regulatory Basis
  - Staff Technical Evaluation and Conclusions
  - Regulatory Findings

- PSAR Section 3.6, “Systems and Components”
  - Overview and Regulatory Basis
  - Staff Technical Evaluation and Conclusions
  - Regulatory Findings

Overview of PSAR Section 3.1

- The principal design criteria (PDC) for the Hermes reactor are based on the approved topical report KP-TR-003-NP, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor.”

- In this section Kairos identifies the relevant regulations and PDCs for the Hermes reactor, as well as the NRC guidance considered in the design

Regulatory Basis

- 10 CFR Section 50.34(a), “Preliminary safety analysis report”
- 10 CFR 50.35, “Issuance of construction permits”
- 10 CFR 50.40, “Common standards”

Staff Technical Evaluation: Review Process

- The NRC staff used the following guidance in its evaluation of the Hermes design criteria:
  - Relevant parts of NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors”
  - Regulatory Guide (RG) 1.232, “Guidance for Developing Principal Design Criteria for Non-Light Water Reactors,” Revision 0
- The staff review included consideration of the limitations and conditions from the staff SE for KP-TR-003-NP.

Section 3.1 Staff Technical Evaluation

- Limitations and conditions from the staff SE for KP-TR-003-NP:
  1. Key design features of Hermes are consistent with those in KP-TR-003-NP
  2. Manufacturing license scope – not applicable to Hermes
  3. NRC-approved guidance to endorse NEI 18-04, “Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Revision 1
     - Not applicable because Hermes follows the guidance of NUREG-1537 instead of NEI 18-04
  4. Use of the term “safety-related”
     - Hermes uses “safety-related” consistent with 10 CFR 50.2
- Terminology
  - “Safety-related” used instead of “safety significant”
  - “Postulated events” used consistent with NUREG-1537

Section 3.1 Regulatory Findings

- The NRC staff finds the design information is consistent with the guidance in RG 1.232 and the applicable criteria in NUREG-1537.

- The staff concludes that the information in Hermes PSAR Section 3.1 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40, and further information can be reasonably left for the OL application.

Overview of PSAR Section 3.6

- PSAR Section 3.6, “Systems and Components,” describes the design bases for the systems and components required to function for safe reactor operation and shutdown.

- PSAR Section 3.6.1 “General Design Basis Information,” describes the safety functions performed by safety-related SSCs:
  - preventing uncontrolled releases of radionuclides,
  - removal of decay heat following a successful reactor trip and
  - control of reactivity

- PSAR Section 3.6.2 “Classification of Structures, Systems, and Components,” describes how SSCs are classified.

Staff Technical Evaluation: Review Process

- The NRC staff used the following guidance in its evaluation of the Hermes design criteria:
  - Relevant parts of NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors”
  - RG 1.29, “Seismic Design Classification for Nuclear Power Plants,” Revision 6, ML21155A003
  - RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” Revision 2, ML013100305
  - RG 1.87, “Acceptability of ASME Section III, Division 5, High Temperature Reactors,” Revision 2, ML22101A263

- The staff review covered the design bases and safety, seismic, and quality classifications.

Staff Technical Evaluation – Design Basis

- Design and construction codes and standards for fluid systems are consistent with RG 1.143 and acceptable.
- High-level safety functions are met by Hermes design because:
  - Reactor vessel and internals support a coolable core geometry and natural circulation for heat transfer to DHRS
  - DHRS operates to remove heat from the core for at least 72 hours following a postulated event where normal cooling systems are unavailable.
  - TRISO fuel pebbles and the Flibe coolant contain fission products.
  - Reactivity control and shutdown system shuts down the reactor and maintains reactor shutdown after an earthquake event
  - Safety-related portion of the Reactor Building protects the reactor vessel and other safety-related SSCs from natural phenomena

Staff Technical Evaluation – Safety and Seismic

- Safety Classification
  - The safety-related classifications of SSCs in Table 3.6-1 meet the definition requirements of 10 CFR 50.2 with one exception:
    - "integrity of the portions of the reactor coolant boundary relied upon to maintain coolant level above the active core."

- Seismic Classification
  - The NRC staff finds that the safety and seismic classification conforms with the guidance in RG 1.29, because safety related SSCs are assigned correctly to the seismic classification, SDC-3, in accordance with ASCE 43-19.
  - The seismic qualification by analysis and testing will be performed in accordance with Section 8.2 and 8.3 of ASCE 43-19.

Staff Technical Evaluation – Quality

- All safety related SSCs are assigned to a quality-related classification.
  - This conforms to RG 1.29 guidance and is acceptable to NRC staff
- ASME Code, Section III, Division 5, will be used for design and fabrication of safety-related mechanical components
  - ANSI/ANS 15.8-1995 (R2005) quality assurance program used rather than NQA-1
- The methodology of load combinations conforms to RG 1.143 Table 3 “Design Load Combinations.”
  - Design transients loading and the number of cycles will be included with the OL application.
- The staff finds that non-condensable gases in the reactor coolant system would not cause a loss of function and present no undue risk, because the design already includes an inert gas blanket above the Flibe coolant.

Section 3.6 Regulatory Findings

- The NRC staff finds the preliminary design information is consistent with the applicable criteria in NUREG-1537.

- The staff concludes that the information in Hermes PSAR Section 3.6 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40, and further information can be reasonably left for the OL application.

Questions?

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

NRC Staff Review for PSAR Sections 2.1 – 2.4, 3.2, and 3.3

Briefing for the Advisory Committee on Reactor Safeguards

March 23, 2023

By the Division of Engineering and External Hazards,
Office of Nuclear Reactor Regulation

Sections 2.1 – 2.4, 3.2, and 3.3

- Format
  - 2.1: Geography and Demography
  - 2.2: Nearby Industrial, Transportation, and Military Facilities
  - 2.3: Meteorology
  - 2.4: Hydrology
  - 3.2: Meteorological Damage
  - 3.3: Water Damage

Overview of staff review

- Kairos applied for a CP and has not specifically requested approval of detailed design information
- Kairos provided a preliminary design description and a discussion of the relevant design bases (e.g., PDCs)
- NRC staff assessed whether the preliminary design information, including identification of relevant design bases, is sufficient to allow the staff to determine that
  - The information meets the relevant regulations for the issuance of a CP
  - The detailed design information can be left to the OL application

Section 2.1 -2.4 Regulatory Basis

- Common to Sections 2.1 to 2.4
  - 10 CFR 50.34(a), “Preliminary safety analysis report.”
  - 10 CFR 50.35, “Issuance of construction permits.”
  - 10 CFR 50.40, “Common standards.”
  - 10 CFR 100.10, “Factors to be considered when evaluating sites.”
  - **Relevant guidance:** NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 2, “Standard Review Plan and Acceptance Criteria.”
- Section 2.1 only:
  - 10 CFR 100.11(a), “Determination of exclusion area, low population zone, and population center distance”
  - NRC Regulatory Guide (RG) 2.6, “Emergency Planning for Research and Test Reactors,” Revision 2.
  - American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.16-2015, “Emergency Planning for Research Reactors.”
- Section 2.2 only:
  - NRC RG 1.91, “Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants,” Revision 2.

PSAR Section 2.1: Geography and Demography

- Site located on previous K-31 and K-33 (Oak Ridge Reservation Gas Diffusion Plant to enrich uranium)
- Site within 0.5 mi is flat
- Exclusion Area Boundary (EAB) coincides with site boundary
- Low Population Zone (LPZ) 0.5 mi
- Emergency Planning Zone (EPZ) boundary set to coincide with site boundary
- Nearest population center is the City of Oak Ridge (2020 Census: 31,402)
- Population projection till 2031 based on Boyd Center for Business and Economic Research, Tennessee

PSAR Section 2.2: Nearby Industrial, Transportation, and Military Installations

- Used Site Safety Analysis Report (SSAR) of the Early Site Permit (ESP) application of the Clinch River Nuclear Site
  - Except for flight operations at Future Oak Ridge Airport
- Audit: Total 5 questions

Sections 2.1 and 2.2 Findings

- Sources of potential hazards (detonation, flammable vapor clouds, toxic chemicals, and fires) from nearby pipelines, highways, railways, waterways, and facilities assessed (PSAR Section 2.2)
  - Distance from the site/facility
  - Quantity of hazardous materials released
  - Potential consequences
- Main Control Room equipped with Chlorine and Ammonia detectors
- Potential aircraft crashes from Oak Ridge Airport significantly high
  - Safety-related portion of Reactor Building will be designed for a crashing small General Aviation aircraft
- Information in PSAR Section 2.1 (Geography and Demography) and 2.2 (Nearby Industrial, Military, and Transportation Facilities) is sufficient and meets the applicable guidance and regulatory requirements to issue construction permit in accordance with 10 CFR 50.35

Section 2.3 Meteorology

Regional Climatology and Local Meteorology

The review of these sections included:

- Thunderstorms, Hail, Lightening
- Extreme Weather Conditions
- Hurricanes, Tornadoes, Winter Precipitation Events
- Wind, Temperature, Precipitation, Atmospheric Stability

NRC staff reviewed:

- The description of the general climate of the region and meteorological conditions relevant to the design and operation of the facility
- The data resources and analytical approaches used by the applicant to prepare the information

NRC staff concludes that the site characteristics associated with the regional climatology and local meteorology are representative of the region of the proposed site and therefore acceptable for use in evaluating the conditions at the proposed site.

Section 2.3 Meteorology

Meteorological Monitoring Program and Atmospheric Dispersion Modeling

The review of these sections included:

- The description of the location and measurements taken at the network of meteorological towers
- The methodology used for atmospheric dispersion modeling

NRC staff reviewed:

- The information describing the network of meteorological towers and their measurement capabilities
- The data taken from the towers to support dispersion analyses at the proposed site.
- The methodology, inputs, and assumptions used in the short term atmospheric dispersion analysis.
- (Kairos states that modeling for routine releases will be provided in the OL application)

NRC staff concludes that:

- Kairos' meteorological monitoring program is acceptable
- The meteorological data set provided is representative of the proposed site and sufficient to support dispersion analyses
- Kairos' short term atmospheric dispersion analysis and calculated dispersion factors for the EAB and LPZ boundary are acceptable

Section 2.3 Regulatory Findings

- NRC staff concludes, based on the review of the information provided in the application, subsequent RAI responses, and the staff's technical evaluation, that the information on meteorology in Hermes PSAR Section 2.3 is sufficient and meets the applicable guidance and regulatory requirements for the issuance of a construction permit in accordance with 10 CFR 50.35.

- Further information on meteorology, namely the details regarding long-term dispersion modeling, can reasonably be left for later consideration in the OL application.

Staff Evaluation of Hydrology

- The applicant provided information on the following topics:
  - Hydrological Description and Flood Record (Hydrologic Setting)
  - Design-Basis Flood Elevation Based on Industry Accepted Methods
  - Hypothetical Dam Failures Scenarios Using Information from TVA
  - Groundwater Levels and Mitigating Measures Based on the Following:
    - The groundwater levels are about 10 feet below grade.
    - There is no groundwater extraction and injection to operate Hermes.
    - The leak of reactor coolant is solidified and confined in the plant building.
    - The fission products, such as tritium, are monitored and confined in the plant.
    - An environmental monitoring program will be implemented to detect any radiological releases beyond normal operational releases.

Hydrology Technical Evaluation Conclusions

- The staff finds:
  - The Hermes site elevation provides approximately 5 feet safety margin above the design-basis flood elevation for the stream and river flood.
  - The design-basis flood elevation for the stream and river flood is higher than FEMA's 500-year flood elevation by approximately 10 feet.
  - The site presents no significant risk to the Hermes facilities due to the postulated flood events.
  - The facility design bases include mitigation and prevention of uncontrolled leakage or loss of reactor coolant to groundwater and surface water.
- Based on above findings, the site hydrology demonstrates that the Hermes site is an adequate site to support facility design bases and satisfies the applicable acceptance criteria of NUREG-1537, Part 2, Section 2.4.

Regulatory Findings on Hydrology

- The staff finds there is reasonable assurance that the preliminary information conforms with 10 CFR 100.10 (c)(3) and supports 10 CFR 50.34(a)(1)(i) by
  - providing flood hazard analyses and site evaluation factors with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the safety margins for design-basis flood elevation for the site.

- The staff concludes that the information on hydrology in Hermes PSAR Section 2.4 is sufficient and meets the applicable guidance and regulatory requirements identified in this section for the issuance of a construction permit in accordance with 10 CFR 50.35, and additional information on hydrology can reasonably be left for later consideration in the OL application.

PSAR Sections 3.2 and 3.3 Meteorological and Water Damage Overview

- Addresses safety-related structures designed to cope with meteorological damage and water damage, including internal and external floods
- Meteorological loading determined in accordance with ASCE/SEI 7-10
- External flood levels based on reviewed data in Chapter 2

Section 3.2 and 3.3 Regulatory Basis

- 10 CFR 50.34(a), “Preliminary safety analysis report”
- 10 CFR 50.35 “Issuance of construction permits”
- 10 CFR 50.40, “Common standards”

- PDC 2 – “Design bases for protection against natural phenomena” which has been approved by the staff (KP-TR-003-NP-A)

- **Relevant Guidance:** NUREG-1537, Part 2, Section 3.2, “Meteorological Damage” and Section 3.3, “Water Damage”

Staff Evaluation - Meteorological

- Appropriate structural loads will be determined in accordance with ASCE/SEI 7-10
  - Tornado and Hurricane wind loads are determined in accordance with RG 1.76 and RG 1.221, respectively
- Design criteria properly align with approved data and predictions in Chapter 2
- Safety-related portion of reactor building designed in accordance with ACI 349 and AISC N690
- Credited to meet PDC 2

Staff Evaluation - Water

- External flood level below structure

- Drainage and grading will be designed to preclude loads from precipitation
  - Staff will review site features during the operating license review
- Internal flooding will be controlled via SSC layout, limiting water volumes, and curbs and drains
  - Staff will review specific details of layout and water volumes during the operating license review

- Credited to meet PDC 2

Technical Evaluation Conclusions

- Safety-related portions of the Reactor Building are designed to appropriate codes and standards
- Meteorological data is appropriately referenced from Chapter 2
- Specific design details that will be reviewed during the operating license have been properly identified

Section 3.2 and 3.3 Regulatory Findings

- The staff finds PSAR Section 3.2 and 3.3 meets the staff guidance provided in NUREG-1537, Part 2, Sections 3.2 and Section 3.3

- The staff finds there is reasonable assurance that the preliminary information conforms with 10 CFR 50.34(a)(3)(iii) by providing information relative to the materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design basis

- The staff concludes the information in Hermes PSAR Section 3.2 and 3.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this section for the issuance of a construction permit in accordance with 10 CFR 50.35 and further information can be reasonably left for the OL application

NRC Staff Review for PSAR Sections 2.5, 3.4, and 3.5

Briefing for the Advisory Committee on Reactor Safeguards

March 23, 2023

By the Division of Engineering and External Hazards,
Office of Nuclear Reactor Regulation

Sections 2.5, 3.4 and 3.5

- Format
  - 2.5: Geology, Seismology, and Geotechnical Engineering
  - 3.4: Seismic Damage
  - 3.5: Reactor Building Structure

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Overview of staff review

- Kairos applied for a CP and has not specifically requested approval of detailed design information
- Kairos provided a preliminary design description and a discussion of the relevant design bases (e.g., PDCs)
- NRC staff assessed whether the preliminary design information, including identification of relevant design bases, is sufficient to allow the staff to determine that
  - The information meets the relevant regulations for the issuance of a CP
  - The detailed design information can be left to the OL application

Regulatory Basis

- 10 CFR 50.34(a), "Preliminary safety analysis report."
- 10 CFR 50.35, "Issuance of construction permits."
- 10 CFR 50.40, "Common standards."
- <u>Relevant guidance:</u> NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 2, "Standard Review Plan and Acceptance Criteria."
- **<u>Section 2.5 only</u>**: 10 CFR 100.10, "Factors to be considered when evaluating sites,"
  - paragraph (c), which requires that in considering the acceptability of a site for a test reactor, physical characteristics of the site, including seismology and geology, should be considered by the NRC.
- **<u>Section 3.5 only</u>** – principal design criteria (PDC):
  - PDC 1 – "Quality standards and records"
  - PDC 2 – "Design bases for protection against natural phenomena"
  - PDC 3 – "Fire protection"
  - PDC 75 – "Reactor building design basis"
  - PDC 76 – "Provisions for periodic reactor building inspection"

PSAR 2.5.1 and 2.5.2 Regional and Site Geology

- Regional geology incorporated by reference from Clinch River ESP 3.5 miles away
- Site underlain by Mascot Dolomite, Murfreesboro Limestone, and Pond Springs Formation within the Appalachian Valley and Ridge Physiographic province
- Foundation rock units are susceptible to karst
  - No surface indications of sinkholes
  - Evidence of karstic activity in borings

Staff Evaluation - Geology

- Characterization of the local and regional geology is adequate
- Potential for surface faulting deferred to operating license.
- **Permit condition:** Removal of overburden soils and weathered rock and geologic mapping will ensure no evidence of karstic dissolution
  – Kairos shall perform detailed geologic mapping of excavations for safety-related engineered structures; examine and evaluate geologic features discovered in those excavations; and notify the Director of the Office of Nuclear Reactor Regulation, or the Director's designee, as specified in 10 CFR 50.4, once excavations for safety-related structures are open for examination by NRC staff.

PSAR Section 2.5.3 Vibratory Ground Motion

- Adopted Clinch River site ground motion hazard curves for site-specific Design Response Spectrum (DRS)
  - 3.5 miles between the two sites
  - Rock site with similar geology
  - Based on NRC-approved seismic source model (NUREG-2115)
  - Based on previously NRC-approved CEUS ground motion model EPRI (2013)
  - Increased Clinch River hazard curves by scale factors of 1.5 to 1.7 to account for use of older ground motion model and potential impact of site-specific site response analysis

Staff Evaluation – Vibratory Ground Motion

- Kairos' use of the Clinch River hazard curves for the site-specific DRS for the Hermes test reactor is appropriate at the CP stage

- For OL Kairos will update its site-specific DRS to incorporate new NRC-approved CEUS ground motion model NGA-East and site amplification factors determined from the results of site response analysis
  - Staff will perform confirmatory evaluation using NGA-East model and local site amplifications

PSAR Section 2.5.4 – Geotechnical Engineering

- Subsurface stratigraphy and material properties determined from boreholes and trenches
- Liquefaction not a concern for safety-related structures
- Bearing capacity of foundation rock expected to be adequate
- Settlement for safety-related structures is negligible – settlement of non-safety related structures controlled through engineered backfill

Staff Evaluation – Geotechnical Engineering

- Liquefaction not a concern for rock units encountered at the site - liquefaction potential for non-safety related structures on engineered fill deferred to OL
- Over-excavation to foundation depth where no evidence of karstic dissolution encountered.
- Additional characterization of foundation bedrock will be addressed in OL

Section 2.5 Regulatory Findings

- NRC staff concludes, based on the review of the information provided in the application, subsequent RAI responses, and the staff's technical evaluation, that the information on geology, seismology, and geotechnical engineering characteristics of the Hermes site in Hermes PSAR Section 2.5 is sufficient and meets the applicable guidance and regulatory requirements for the issuance of a construction permit in accordance with 10 CFR 50.35.

- Further information on geology, seismology, and geotechnical engineering can reasonably be left for later consideration since this information is not necessary to be provided as part of a CP application.

PSAR Section 3.4 Seismic Design Overview

- Addresses SSCs required to remain functional after an earthquake (ASCE 43-19 seismic design category 3)
- Uses graded approach from ASCE 43-19
- DRS based on seismic design category 3 and site hazard from PSAR Section 2.5
- Seismic response analysis and Soil Structure Interaction will be performed in accordance with ASCE 4-16

Staff Evaluation – Seismic Design

- Safety-related portions of the Reactor Building will be designed to appropriate seismic codes and standards (ASCE 43-19, ASCE 4-16)

- DRS is properly developed based on site data from Chapter 2

- Specific design details (seismic model, etc.) that will be reviewed during the operating license have been properly identified

- Reasonable level of detail and information provided for issuance of a CP

PSAR Section 3.5 Plant Structures Overview

- Describes principal structural elements and design of the reactor building (RB)
- Summarizes how reactor building meets PDCs 1, 2, 3, 75 and 76
  - Supports or repeats information contained in PSAR Sections 3.2, 3.3, and 3.4
- Discusses Seismic Isolation System

Staff Evaluation – PDC 1 and 2

- PDC 1 –SSCs important to safety shall be designed to quality standards
  - Safety-related SSCs will be designed and constructed in accordance with ACI 349 and AISC N690

- PDC 2 – SSCs important to safety shall be designed to withstand natural phenomena
  - Structures will be designed to withstand appropriate natural loads (see SE Sections 3.2, 3.3, and 3.4)
  - Structures designed in accordance with appropriate nuclear codes (ACI 349 and AISC N690)

Staff Evaluation – PDC 3, 75 and 76

- PDC 3 – SSCs important to safety shall be designed and located to minimize effects of fires
  - RB uses low combustible materials and physically separates SSCs
  - Fire protection program provided (SE Section 9.4)
- PDC 75 – RB shall protect the geometry of the decay heat removal system (DHRS) from natural phenomena
  - DHRS is located inside safety-related portion of RB, which provides assurance the DHRS will be protected
- PDC 76 – RB shall be designed to permit periodic inspection
  - The RB, including the seismic isolation system, is designed to allow access and permit inspection

Staff Evaluation – Seismic Isolation

- Safety-related portion of RB will be seismically isolated
  - Minimal information provided on design
- ‘Moat’ separates seismically isolated portion from rest of plant
- System will be accessible and inspectable throughout life of plant
- Staff verified system will be designed in accordance with Chapter 9 of ASCE 43-19 and details of system and analysis will be provided in the Operating License application

Technical Evaluation Conclusions

- Safety-related portions of the RB will be designed to appropriate nuclear codes and support PDCs 1, 2, 3, 75 & 76

- The seismic isolation system will be designed to the appropriate guidance in ASCE 43-19

- Specific design details that will be reviewed during the operating license have been properly identified

Sections 3.4 and 3.5 Regulatory Findings

- The staff finds there is reasonable assurance that the preliminary information is consistent with the applicable criteria in NUREG-1537 and the relevant PDC and conforms with 10 CFR 50.34(a)(4) by providing a preliminary analysis and evaluation of the design and performance of SSCs

- The staff concludes the information in Hermes PSAR Sections 3.4 and 3.5 is sufficient and meets the applicable guidance and regulatory requirements identified in this section for the issuance of a construction permit in accordance with 10 CFR 50.35 and further information can be reasonably left for the OL application

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NRC Staff Review for PSAR Section 4.2 Reactor Core

Briefing for the Advisory Committee on Reactor Safeguards

March 23, 2023

By the Division of Advanced Reactors and Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Agenda

- PSAR Section 4.2.1, “Reactor Fuel”
  - Overview and Regulatory Basis
  - Staff Technical Evaluation, Conclusions, and Findings
- PSAR Section 4.2.2, “Reactivity Control and Shutdown System”
  - Overview and Regulatory Basis
  - Staff Technical Evaluation, Conclusions, and Findings
- PSAR Section 4.2.3, “Neutron Startup Source”
  - Overview and Regulatory Basis
  - Staff Technical Evaluation, Conclusions, and Findings

Overview of PSAR Section 4.2.1

- The TRISO particle is the primary fission product barrier
- Uses the larger AGR-2 fuel kernel
- Uses nominal AGR program coating thicknesses
- The TRISO particles are arranged in an annulus near the pebble outer edge
- Smaller than typical HTGR pebble size
- Covers unfueled moderator pebbles
- Covers fuel performance monitoring

Section 4.2.1 Regulatory Basis

- 10 CFR 50.34(a), "Preliminary safety analysis report."
- 10 CFR 50.35, "Issuance of construction permits."
- 10 CFR 50.40, "Common standards."
- **Relevant guidance:** NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 2, "Standard Review Plan and Acceptance Criteria."

- Principal design criteria (PDC):
  - PDC 10 – "Reactor design"
  - PDC 16 – "Containment Design"
  - PDC 34 – "Residual heat removal"
  - PDC 35 – "Passive residual heat removal"

Referenced Topical Reports

- KP-TR-003-NP-A, Revision 1, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor”
- EPRI-AR-1-NP-A, “Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance”
  - ➢ Used to evaluate TRISO particle parameters and performance envelope
- KP-TR-011, Revision 2, “Fuel Qualification Methodology for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR)”
  - ➢ Used to evaluate expected pebble behavior
- KP-TR-010-NP-A, Revision 3, “KP-FHR Fuel Performance Methodology”
  - ➢ Illustrative examples for postulated event fuel failure fractions and incremental failure probability to support reasonable assurance finding
  - ➢ The KP-BISON code has not been approved by the staff

Staff Evaluation – TRISO Particles

- Particles will be within the measured UCO particle parameters of EPRI-AR-1, Table 5-5
- AGR-2 irradiated conditions bound the Hermes expected normal operation
- Expected postulated event peak particle temperatures are below 1600 ^{o}C, the lowest AGR program safety testing temperature
- AGR test program did not include overpower transient tests (discussed on a following slide)

Staff Evaluation – Pebbles

- Fuel Pebble
  - Not credited for fission product retention
  - The fuel qualification topical report, KP-TR-018, addresses pebble testing to ensure protection of the TRISO particles and buoyance function
  - Specific correlations, if necessary, will be derived from the test data and applied in the Hermes FSAR

- Moderator Pebbles
  - Contains no fuel and is a nonsafety-related component
  - Same material as the fuel pebble matrix
  - Testing for buoyancy, wear, strength, and salt ingress will be same as the fuel pebble
  - Will be inspected by the PHSS like fuel pebbles
  - Staff review focused on the potential impact to safety-related functions like decay heat removal

Staff Evaluation – Overpower Transient

- AGR safety test program consisted of relative slow heats like those experienced in loss of flow heat-up events
  - Hermes overpower transient heat-up rates are significantly faster

- Overpower events can potentially lead to fuel melt or non-melt mechanical failures

- Limiting overpower transients are generally rod ejection and rapid element withdrawals
  - Rod ejection is precluded by design due to the low differential pressure

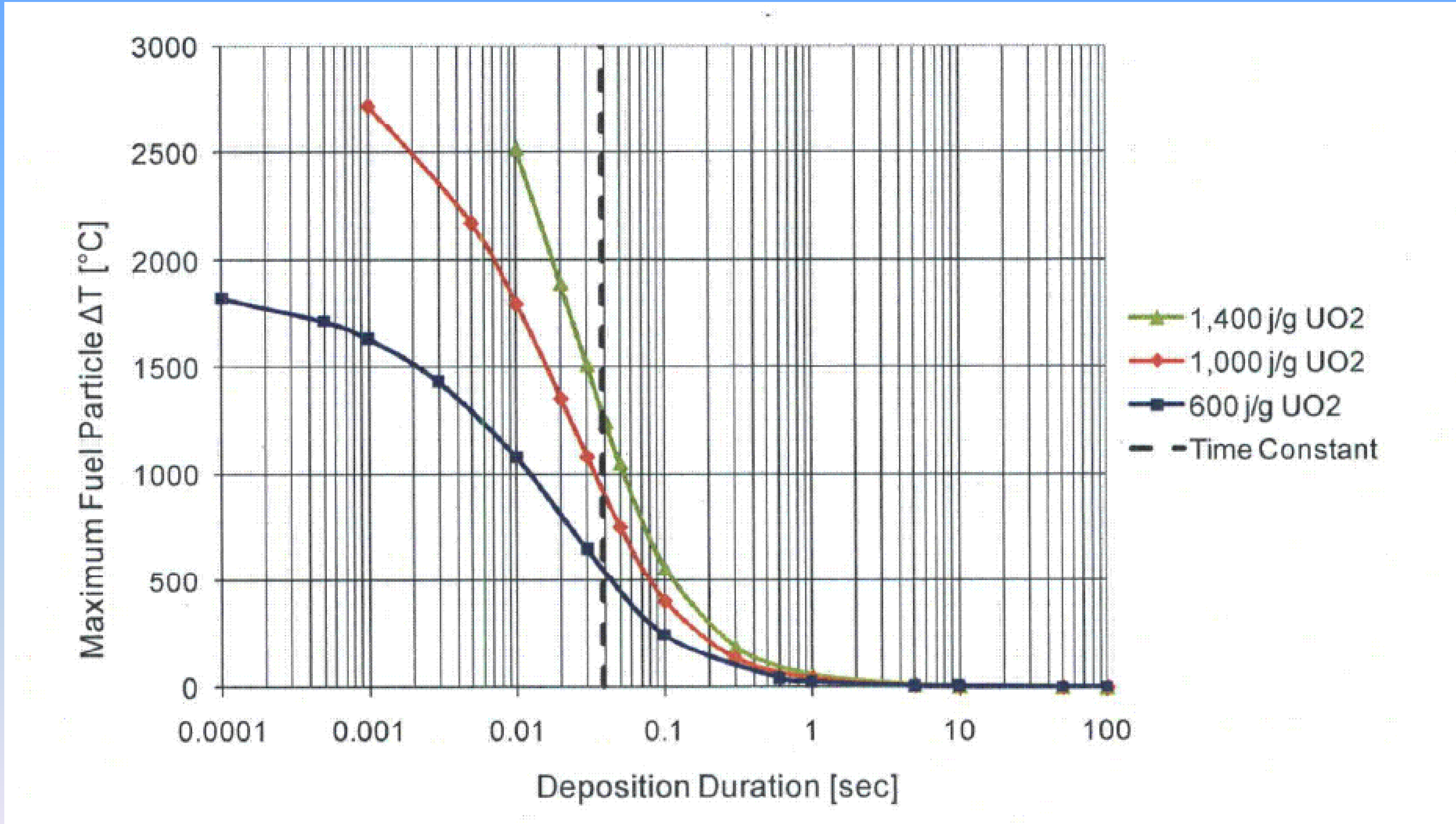
- Staff evaluated the maximum control element withdrawal presented in Figure A1-2 of Postulated Events Analysis Methodology technical report KP-TR-018

Staff Evaluation – Overpower Transient

- The maximum control element withdrawal kernel temperature is approximately 1,080 °C
  - Below the unirradiated $UC_{1.96}$ and $UC_{1.86}$ melt temperature (1,820 °C and 2,350 °C, respectively)
    - UO_2, UC and $UC_{1.86}$ were the reported AGR phases
  - The energy deposition is low compared to the 1,400 J/g-UO_2 failure threshold

- Overpower event non-melt failures are also expected to be negligible
  - The time to reactor trip is approximately 10 seconds
  - The fuel thermal time constant is between 30 to 300 milliseconds
  - For events lasting on the order of greater than 1 second the maximum fuel particle differential temperature is near zero and is independent of the energy deposition

Staff Evaluation – Overpower Transient



Staff Evaluation – Overpower Transient

- Negligible incremental failures predicted by the KP-BISON code for the maximum control element withdrawal analysis in KP-TR-018
  - The KP-BISON code has not been approved but provides insights to the expected in-service incremental failure fraction
  - The predicted in-service incremental failure fraction using a verified KP-BISON code will be performed as part of the OL application review

- Based on the predicted low, maximum TRISO kernel temperature and the corresponding margin to melt, the low stress caused by the small differential coating temperatures, and review of the preliminary KP-BISON results, the staff finds that incremental failures are not expected during the worst case overpower postulated event.

Staff Evaluation – Fuel Performance Monitoring

- Three non-destructive means are available to monitor fuel performance:
  - Cover gas and reactor coolant monitoring will be performed, which provides an indication of particle failures
    - A technical specification (TS) on coolant activity is given in PSAR Chapter 14 but specific values will be provided as part of the OL
  - Fuel pebbles will be examined for damage and burnup by the pebble handling system
    - Pebbles which show abnormal wear, cracking or missing surfaces will be removed from service
    - Pebbles will be discharged to prevent exceeding the burnup limit

Staff Evaluation – Fuel Performance Monitoring

- Destructive testing of the Hermes fuel will be performed as stated in Section 3.9.3 of KP-TR-011
    - The destructive testing will determine failed fuel fractions, pebble wear and the extent, if any, of Flibe ingress
- The destructive testing can provide additional fuel performance code validation data and provide input to revise any future KP-FHR fuel PIRT

4.2.1 Technical Evaluation Conclusions

- The TRISO particles are expected to operate within the bounds as defined by the AGR-2 test data
- Incremental failures are expected to be negligible based on assessing the worst overpower postulated event and the AGR program safety testing which addresses the slower heat-up events
- Fuel and moderator pebble testing programs are sufficient to develop the relevant acceptance criteria or correlations to ensure the pebble safety functions are satisfied
- Fuel monitoring is adequate to determine unexpected fuel failures

Section 4.2.2 Regulatory Basis

- 10 CFR 50.34(a), “Preliminary safety analysis report.”
- 10 CFR 50.35, “Issuance of construction permits.”
- 10 CFR 50.40, “Common standards.”
- **Relevant guidance:** NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 2, “Standard Review Plan and Acceptance Criteria.”

- Principal design criteria (PDC):
  - PDC 2 – “Design bases for protection against natural phenomena”
  - PDC 4 – “Environmental and dynamic effects design bases”
  - PDC 23 – “Protection system failure modes”
  - PDC 26 – “Reactivity control systems”
  - PDC 28 – “Reactivity limits”
  - PDC 29 – “Protection against anticipated operational occurrences”

Section 4.2.2 Reactivity Control and Shutdown System Overview

- The reactivity control and shutdown system (RCSS) is credited with shutdown
  - The RCSS ensures safe shutdown by inserting and withdrawing elements to control reactivity during normal operation and in response to abnormal conditions (or postulated events)
- 4 control and 3 shutdown elements use boron carbide (B_4C) absorber material
- Control elements (CEs)
  - Insert into the graphite reflector
  - Range of insertion positions
- Shutdown elements (SEs)
  - Insert directly into the pebble bed
  - Either fully withdrawn or fully inserted
  - Credited with shutting down the reactor during postulated events

Section 4.2.2 Reactivity Control and Shutdown System Overview

- Control and shutdown elements are controlled using counter-weighted winch systems
  - A wire-rope is connected to the element and travels up around the winch drum (also known as a sheeve) and down to a counter-weight
  - The winch drum is rotated by an electric motor

- There is an electric clutch between the winch drum and the electric motor to control the element movement
  - During a reactor trip, the electric clutch opens, allowing the winch drum to rotate freely, and the elements are released from their drives, allowing them to drop as a result of gravity.
  - The release of the clutch for the shutdown elements is the primary safety-related reactor trip mechanism

Staff Evaluation – PDCs 2 and 4

- PDC 2, "Design bases for protection against natural phenomena"
  - Kairos will perform a one-time test before operation that deflects SE guide structures by a maximum misalignment that would be caused by a design basis (DB) earthquake to confirm that element insertion time is bounded by the insertion time assumed in the Ch. 13 analysis

- PDC 4, "Environmental and dynamic effects design bases"
  - Kairos will perform testing prior to operation for SE wear during movement
  - Kairos will perform analyses for SE internal gas release and swelling of B4C and for SE stress
  - Kairos will perform analyses for SEs and CEs to show they meet American Society of Mechanical Engineers standards

Staff Evaluation – PDCs 23, 26, 28, and 29

- PDC 23, "Protection system failure modes"
  - SEs accomplish safe shutdown via gravity insertion on a reactor trip signal or on a loss of normal electrical power
  - Removal or loss of power causes electromagnetic clutch to open
- PDC 26, "Reactivity control systems"
  - PDC 26 is discussed in Section 4.5
- PDC 28, "Reactivity limits"
  - NRC scoping calculations predicted that TRISO fuel maintains integrity during postulated insertion of excess reactivity events
  - Rod ejection is discussed in Chapter 13
- PDC 29, "Protection against anticipated operational occurrences"
  - Kairos will perform testing (as discussed in PDCs 2 and 4)
  - Kairos will perform periodic inspection of SEs and coolant to look for evidence of SE damage or failure

4.2.2 Technical Evaluation Conclusions

- The NRC staff finds the preliminary design information provided is consistent with PDC 2, 4, 23, 26, 28, and 29 and is consistent with the relevant acceptance criteria of NUREG-1537.

- Staff has reasonable assurance that the RCSS will perform its safety functions of reactivity control and shutdown.

4.2.3 Neutron Source Overview, Staff Evaluation, and Conclusions

- **<u>Relevant guidance:</u>** NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 2, “Standard Review Plan and Acceptance Criteria.”

- Overview
  - PSAR does not specify source type
  - Neutron startup source is placed in a metal sheath and located in reflector

- The staff finds this meets applicable NUREG-1537 acceptance criteria because it should be capable of performing in its environment and is removable and replaceable.

Section 4.2 Regulatory Findings

- The NRC staff concludes that the preliminary design provided in PSAR Section 4.2 is consistent with the applicable PDCs and acceptance criteria in NUREG-1537.

- The NRC staff finds that the information in Hermes PSAR Section 4.2 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40, and further information can be reasonably left for the OL application

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NRC Staff Review for PSAR Section 4.5 Nuclear Design

Briefing for the Advisory Committee on Reactor Safeguards

March 23, 2023

By the Division of Advanced Reactors and Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Agenda

- Overview of PSAR Section 4.5, “Nuclear Design”
- Regulatory basis and PDCs
- Referenced topical reports
- Staff technical evaluation
- Technical conclusions
- Regulatory Findings

Regulatory Basis

- 10 CFR 50.34(a), “Preliminary safety analysis report.”
- 10 CFR 50.35, “Issuance of construction permits.”
- 10 CFR 50.40, “Common standards.”
- **Relevant guidance:** NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 2, “Standard Review Plan and Acceptance Criteria.”

- Principal design criteria (PDC):
  - PDC 10 – “Reactor design”
  - PDC 11 – “Reactor inherent protection”
  - PDC 12 – “Suppression of reactor power oscillations”
  - PDC 26 – “Reactivity control systems”

Referenced Topical Reports

- KP-TR-003-NP-A, Revision 1, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor”

- KP-TR-013-NP, Revision 4, “Metallic Materials Qualification for the Kairos Power Testing Program”
  - Used to evaluate vessel irradiation

- KP-TR-017-P, Revision 1, “KP-FHR Core Design and Analysis Methodology” (technical report)

Nuclear Design Methodology

- Two main physics codes:
  - Serpent 2 for neutron/gamma transport and isotopics
    - Continuous energy (CE) Monte Carlo method
    - Doppler-broadened cross section data provided a priori by KACEGEN
  - STAR-CCM+ for pebble tracking and thermal-fluidics (T/F)
    - Discrete element method (DEM) for pebble flow
    - Porous media approach for fluid and structure temperatures

- Used within two analysis sequences:
  - KPATH (coupled transport/T-F) – iterations between Serpent and STAR-CCM+ to update temperature fields based on power distribution and vice versa
  - KPACS (fuel cycle analysis) – updates isotopics within geometry as core operates based on DEM-calculated pebble flow fields

Staff Evaluation – Methodology

- Basic methodologies appear sound
  - Temperature-dependent CE cross section libraries generated across wide range, fine grid
  - KPACS accounts for radial and axial zoning, local effects on neutron spectrum, differential pebble depletion per pass in core
  - KPATH provides for thermal feedback in the neutronics solution
  - Models provide robust means to calculate safety parameters (shutdown margin, reactivity coefficients, delayed neutron and decay heat data)
  - Coolant temperature feedback calculated assuming limiting isotopic composition

Remaining Methodology Items for Operating License

- PSAR analysis mainly focuses on *nominal* core performance
- Neutronics uncertainties have not been formally established
  - Additional work is planned to establish conservatism in the presently assumed uncertainties in key figures of merit
  - Submittal expected to support operating license (OL)
- No validation or assessment has been submitted for review
  - KP-TR-017-P includes brief description of validation framework for relevant STAR-CCM+ and Serpent models to support OL
- Additional details on certain modeling approaches (e.g., shutdown control elements when inserted in bed, reactor vessel fluence, operation with control elements partially inserted) are expected
- These codes and methods are not considered "reviewed and approved" at this stage

Staff Scoping Analysis of Hermes

- NRC developed several ‘representative’ non-LWR systems models since 2020
  - Part of “Non-LWR Vision and Strategy, Volume 3” covering severe accidents/source term
  - Included UC Berkeley Mark 1 design, representing TRISO pebble fueled/molten salt cooled FHR
  - SCALE code suite used for inventory and reactor physics data generation (ORNL)
  - MELCOR used for accident progression using SCALE-produced data (Sandia)
  - FHR demonstration public workshop 9/2021 (ML21256A231)

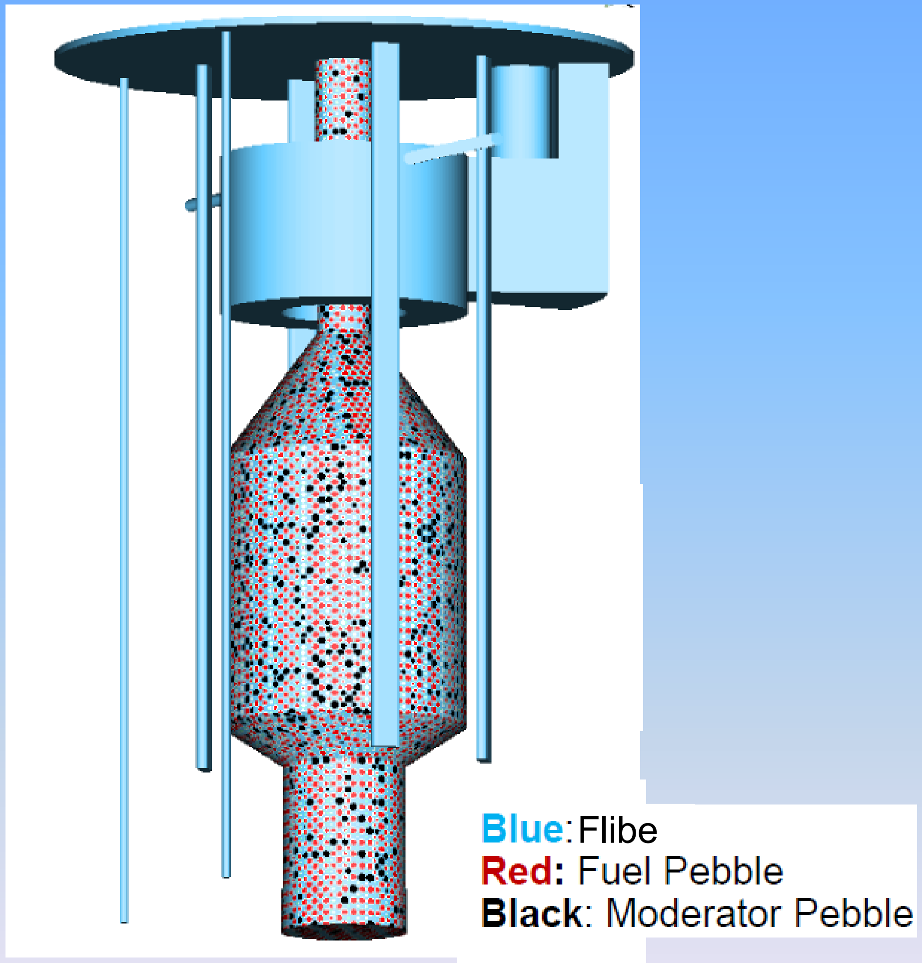
- These models/workflow were rapidly adapted to support Hermes scoping-level analysis (early 2022)

SCALE Analysis Approach

- This is a *scoping* rather than a *confirmatory* analysis
  - Rapid turnaround to support review timeline (initiated in January, presented to NRR staff in March 2022)
  - Exclusive reliance on non-proprietary data from Hermes PSAR
  - Inform review staff's engineering judgement
- SCALE FHR workflow described in ML22152A163
  - Justification of multigroup energy treatment
  - Generation of equilibrium isotopic inventory using 2D slice models of reactor
  - 3D full core calculations for reactor physics data
    - Eigenvalue
    - Power distribution
    - Feedback coefficients
    - Kinetics parameters

SCALE Model Description

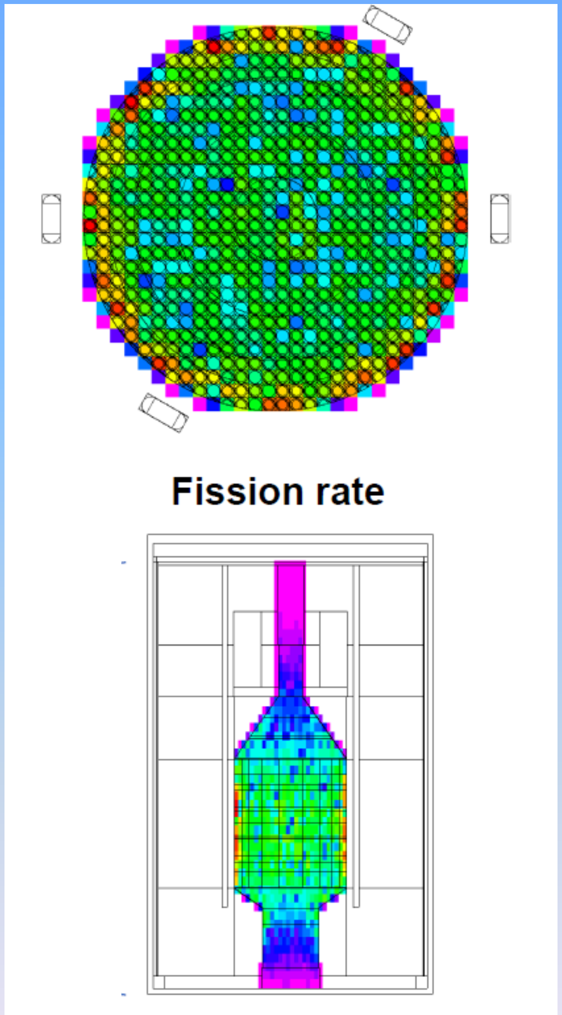
- Multigroup Monte Carlo transport using KENO-VI, isotopics calculated with ORIGEN
- Random pebble geometry approximated by regular lattice
- Equilibrium isotopics generated iteratively via 2D slice models with SCALE/TRITON
- Axially-dependent fuel isotopics inserted into 3D core model for reactivity and power shape evaluations
- Does not currently include shutdown (in-bed) elements – on list for further development



SCALE Results: Power Distribution

Relative Power	Kairos PSAR	SCALE
Axial (-)	1.2	1.19
Radial (-)	1.2	1.76
Peak Pebble (-)	1.8	2.09

- Excellent agreement axially
- Not as good radially or at peak pebble
  - Inconsistencies in peaking definitions
  - Inconsistencies in reflector model
  - Differences in pebble treatment – averaged vs. pass/location dependent isotopics
- *Reasonable*, given uncertainties – could be resolved with additional information



SCALE Results: Reactivity Coefficients

Parameter	Kairos PSAR	SCALE*
Fuel Doppler (pcm/K)†	-4.1	-4.30 ± 0.27
Moderator (pcm/K)†	-0.4	-0.47 ± 0.13
Coolant (pcm/K)†	-1.6	-1.62 ± 0.02
Void (pcm/% void, @3% void)	-53	-46.6 ± 4.0
Reflector (pcm/K)†	+2.0	+1.92 ± 0.23
β_{eff} (pcm)	605	576 ± 10

* - includes Monte Carlo uncertainty

† - calculated assuming temperature distributions provided by MELCOR

Staff Evaluation – Power Distribution

- Codes
  - Serpent 2 is used to calculate core power distribution
  - STAR-CCM+ is used to calculate material temperature distributions
  - Serpent 2 and STAR-CCM+ are coupled and output the distributions
- The core power distribution is characterized by axial peaking factor, radial peaking factor, and total peaking factor
  - Peaking factors are calculated using Serpent 2 by tallying fission power in each pebble and dividing by the average power per pebble in the core
- Staff scoping calculations showed reasonable agreement with Kairos' preliminary calculations of power distribution and total pebble peaking factors
  - The core power distributions are an input to the fuel performance calculations. The staff finds that this is consistent with PDC 10.

Staff Evaluation – Shutdown Margin

- Shutdown margin (SDM) design criteria is $k_{eff} < 0.99$
  - SDM is defined relative to the margin to Flibe freezing temperature in the core
  - SDM is maintained at all core states

- Control element worth is calculated from changes in k_{eff} resulting from perturbing element axial positions in Serpent 2.
  - The single most reactive element is assumed to be fully withdrawn from the core.

- Kairos will perform source range control element worth testing

- Staff finds this is consistent with PDC 26

Staff Evaluation – Reactivity Coefficients

- Reactivity coefficients were calculated using Serpent 2
- Reactivity coefficients include
  - Fuel temperature (-)
  - Moderator temperature (-)
  - Coolant temperature (-)
  - Coolant void (-)
  - Reflector temperature (+)
- Reflector temperature reactivity coefficient is slightly positive, but thermal expansion in the reflector, which is a negative feedback, was conservatively ignored
- Moderator temperature reactivity coefficient includes change in reactivity due to change in temperature in fuel pebble graphite and in graphite pebbles
- Coolant temperature reactivity coefficient accounts for density changes. Coolant void reactivity coefficient is change in reactivity due to change in coolant void fraction.

Staff Evaluation – Reactivity Coefficients

- Overall reactivity coefficient is negative
- Staff scoping calculations showed reasonable agreement with Kairos' preliminary calculations for reactivity coefficients
- Once Hermes achieves criticality and is at zero power, Kairos will perform isothermal reactivity coefficient testing
- Staff finds this is consistent with PDC 11

Staff Evaluation – Vessel Irradiation

- Vessel lifetime is 4 years

- Vessel is shielded by the core barrel, the reflector, and the Flibe coolant

- Serpent 2 is used to calculate fast neutron fluence and alpha generation on the vessel received from the core and pebble insertion and extraction lines

- Kairos' preliminary calculation of displacements per atoms (dpa) in the vessel is within the low-level irradiation value discussed in KP-TR-013-NP

Staff Evaluation – Nuclear Transient Parameters

- Nuclear transient parameters are outputs from Serpent 2 and are used as inputs for transient analyses

- Nuclear parameters include
  - Neutron generation time
  - Delayed neutron fraction groups and their decay constants
  - Prompt neutron lifetime

- Staff's evaluation of nuclear core design limits (i.e., Burnup, peak fuel temperature, peak particle power, and peak fluence) are discussed in SE Section 4.2.1, "Reactor Fuel"

Staff Evaluation – Monitoring

- Neutron flux will be monitored using
  - 4 power range ex-core detectors located in azimuthally symmetric locations outside the reactor vessel at mid-core elevation
  - 4 source range ex-core detectors located in relation to the startup source for best detectability of criticality
- During normal operation, these detectors will be used to monitor core power and the flux rate trip signal
- Source range detectors used during startup
- Gamma spectrometry used to evaluate fuel pebble burnup
  - Staff did not make any findings on Kairos' burnup monitoring plan
- Kairos will perform neutron flux distribution verification tests during startup
- Staff finds this is consistent with PDC 10

Technical Evaluation Conclusions

- Kairos PDC 10, “Reactor design”
- Kairos PDC 11, “Reactor inherent protection”
- Kairos PDC 12, “Suppression of reactor power oscillations”
- Kairos PDC 26, “Reactivity control systems”

- Preliminary design information provided is consistent with PDC 10, 11, 12, and 26 and is consistent with acceptance criteria of NUREG-1537
- Analytical methods for nuclear design are acceptable at this stage of the design

Section 4.5 Regulatory Findings

- The NRC staff concludes that the preliminary design provided in PSAR Section 4.5 is consistent with the applicable PDCs and acceptance criteria in NUREG-1537.

- The NRC staff finds that the information in Hermes PSAR Section 4.5 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40, and further information can be reasonably left for the OL application

Questions?

NRC Staff Review for PSAR Section 4.3 Reactor Vessel System

Briefing for the Advisory Committee on Reactor Safeguards

March 24, 2023

By the Division of Advanced Reactors and Non-Power Production and Utilization Facilities,

Office of Nuclear Reactor Regulation

Agenda

- Overview of PSAR Section 4.3 “Reactor Vessel System”
- Regulatory basis and PDCs
- Referenced topical reports
- Staff technical evaluation
- Technical conclusions
- Regulatory Findings

PSAR Section 4.3 Reactor Vessel System

- Comprised of vessel shell, top head, bottom head, and vessel internals
  - Internals include reflector blocks, fluidic diodes, core barrel, and reflector support structure
- Contains the core, provides for circulation of the reactor coolant and pebbles, and insertion of RCSS elements
- Safety related system

Regulatory Basis

- 10 CFR 50.34(a) “Preliminary safety analysis report”
- 10 CFR 50.35 “Issuance of construction permits”
- 10 CFR 50.40, “Common standards”
- The following NRC staff approved PDC (KP-TR-003-NP-A):
  - PDC 1 – “Quality standards and records”
  - PDC 2 – “Design bases for protection against natural phenomena”
  - PDC 4 – “Environmental and dynamic effects design bases”
  - PDC 10 – “Reactor design”
  - PDC 14 – “Reactor coolant boundary”
  - PDC 30 – “Quality of reactor coolant boundary”
  - PDC 31 – “Fracture prevention of reactor coolant boundary”
  - PDC 32 – “Inspection of reactor coolant boundary”
  - PDC 33 – “Reactor coolant inventory maintenance”
  - PDC 34 – “Residual heat removal”
  - PDC 35 – “Passive residual heat removal”
  - PDC 36 – “Inspection of the passive residual heat removal system”
  - PDC 37 – “Testing of the passive residual heat removal system”
  - PDC 74 – “Reactor vessel and reactor system structural design basis”

Referenced Topical Reports

- KP-TR-003-NP-A, Revision 1, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor”
- KP-TR-013-NP, Revision 4, “Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor,”
- KP-TR-014-NP, Revision 4, “Graphite Material Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor”

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Staff Evaluation – PDC 2

- The reactor vessel, vessel internals, and vessel attachments are classified as Seismic Design Category (SDC)-3 per ASCE 43-19 “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities”.

- The safety related SSCs will be protected from the failure of nearby non-safety related SSCs during a design basis earthquake.

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Staff Evaluation – PDC 4

- The reactor vessel is designed to account for internal and external static and dynamic loads, including static weight, seismic loads, and forces from the pebble bed, coolant, and core components.

- Pipe whip hazards are not a concern because there are no pressurized piping systems near the reactor vessel.

Staff Evaluation – PDC 10

- PSAR states that the vessel and internals needed to define coolant flow path
  - Integrity of SS and graphite components
- The NRC staff has reasonable assurance that the design will be consistent with PDC 10
  - Graphite qualified as per KP-TR-014-NP and designed to Section III Division 5
  - 316H SS qualified as per KP-TR-013-NP and designed to Section III Division 5
  - The reactor vessel, vessel internals, and vessel attachments are classified as SDC-3 per ASCE 43-19 to account for dynamic behaviors to ensure continued functionality during and after a design basis earthquake event.
  - The reactor vessel system is protected from the failure of nearby non-safety related SSCs impacting safety significant SSCs.

Staff Evaluation – PDC 14

- PSAR states
  - Vessel material and weld metal will be qualified consistent with Section III Division 5 and KP-TR-013-NP
  - Designed to accommodate operational and transient stresses
- NRC staff has reasonable assurance the reactor vessel system can be designed consistent with PDC 14
  - Qualification testing of 316H SS and weld filler will demonstrate compatibility between vessel and weld metals with the reactor coolant
  - Catch basins for leak monitoring
  - Inspection and monitoring programs to be reviewed at OL stage
  - Vessel will be monitored and is designed for in-service inspection
  - The load combination methodology conforms to Table 3, “Design Load Combinations,” of RG 1.143.

Staff Evaluation – PDCs 30 and 31

- PSAR states reactor vessel will be fabricated, erected, and tested consistent with Section III Division 5
- NRC staff has reasonable assurance that the design will be consistent with PDCs 30 and 31
  - Temperatures up to 750°C and proposed Safety Limit on vessel temperature
  - KP-TR-013-NP contains additional testing to account for material degradation
  - TR includes extension of weld filler metal qualifications
  - Consistency with NRC-endorsed standards
  - Leakage detection
  - Minimization of air ingress and coolant purity limits

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Staff Evaluation – PDC 32

- PSAR states coupons, component monitoring, and inspection will assess structural integrity and leak-tightness of reactor coolant boundary
- NRC staff has reasonable assurance the reactor vessel system can be designed consistent with PDC 32
  - Preliminary information is consistent with guidance in NUREG-1537 to assess irradiation of vessel materials
  - Staff will evaluate final design to confirm
    - Results of qualification testing in conjunction with final design and associated surveillance, inspection, and performance monitoring programs to determine need for fracture toughness coupons
    - Monitoring and inspection programs will be performed to assure vessel integrity

Staff Evaluation – PDC 33

- The PSAR states that the core barrel is designed to include anti-siphon features to limit reactor coolant inventory loss
- Staff evaluation of PDC 33 and NUREG-1537 guidance is in Chapter 5 of the SE

Staff Evaluation – PDC 34

- PSAR states that vessel internals design supports decay heat removal
- NRC staff has reasonable assurance the reactor vessel system can be designed consistent with PDC 34
  - Design maintains pathway for coolant
    - In conjunction with Primary Heat Transfer System for forced flow in normal operation
    - Via fluidic diodes for natural circ in postulated events
- Many other sections of the SE describe aspects of this PDC
  - Ch 4.6 (thermal hydraulics), Ch 6.3/13 (PHTS), Ch 5 (anti-siphon)

Staff Evaluation – PDCs 35, 36, and 37

- PSAR states that fluidic diode and reactor design maintain a flow path to allow for natural circulation of coolant
- NRC staff has reasonable assurance the reactor vessel system can be designed consistent with PDCs 35, 36, and 37
  - KP-TR-013-NP, Rev 4 testing will bound the design temperature and fluence of diode
  - Normal flow path – temperature monitoring at exit of reactor vessel
  - Postulated accident flow path - inspection of fluidic diode and temperature monitoring fluidic diode pathway

Staff Evaluation – PDC 74

- PSAR states that reflector will be designed to meet Division 5 requirements

- NRC staff has reasonable assurance the reactor vessel system can be designed consistent with PDC 74
  - Preliminary info consistent with NUREG-1537 requirements
    - Design limits - KP-TR-014-NP, Rev 4 includes testing on effects needs to design the reflector (irradiation, thermal, etc.)
    - Surveillance - thermal mapping of reflector
  - RVSS is designed to provide the structural support for the reactor vessel under static and dynamic loadings (e.g., seismic).

Testing and Inspection

- PSAR states design of vessel/internals allows for monitoring, inspection and maintenance
- PSAR states testing and inspection of reactor vessel and internals will be submitted with the OL application
  - Staff will review these programs at that time

Conclusion

- NRC staff finds the preliminary design information is consistent with the applicable criteria in NUREG-1537 and the applicable PDC

- The staff concludes information in Hermes PSAR Section 4.3 is sufficient for the issuance of a CP in accordance with 10 CFR 50.35 and further information can be reasonably left for the OL application

Questions?

NRC Staff Review for PSAR Section 4.4 Biological Shield

Briefing for the Advisory Committee on Reactor Safeguards

March 24, 2023

By the Division of Advanced Reactors and
Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Overview of PSAR 4.4 “Biological Shield”

- Radiation shielding for worker protection during operation
- Reinforced concrete structures
- Safety related component for structural support and external event protection
  - Not safety related for radiation protection function

Staff Evaluation – Biological Shield

- Assess whether the PSAR provides an acceptable basis for the development of the biological shield
- Determine if the objectives of the biological shield design basis are sufficient to protect the health and safety of the public and facility staff
- Assess whether there is reasonable assurance that Kairos will comply with the regulations in 10 CFR Part 20 during Hermes facility operation

Staff Evaluation - Biological Shield Radiation Protection Design Basis

- Audit confirmed that Kairos performed preliminary shielding analysis to support PSAR
  - Isotopic data based on fuel and Flibe sources

- Evaluation of performance to meet Part 20, including shielding analyses, to be provided in OL application

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NRC Staff Review for PSAR Section 4.6 Thermal-Hydraulic Design

Briefing for the Advisory Committee on Reactor Safeguards

March 24, 2023

By the Division of Advanced Reactors and Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Agenda

- Regulatory basis and PDCs
- Referenced topical reports
- Staff technical evaluation
- Technical conclusions
- Regulatory Findings

Regulatory Basis

- 10 CFR 50.34(a), “Preliminary safety analysis report.”
- 10 CFR 50.35, “Issuance of construction permits.”
- 10 CFR 50.40, “Common standards.”

- Principal design criteria (PDC):
  - PDC 10 – “Reactor design”
  - PDC 12 – “Suppression of reactor power oscillations”
  - PDC 34 – “Residual heat removal”
  - PDC 35 – “Passive residual heat removal”

- **Guidance:** NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 2, “Standard Review Plan and Acceptance Criteria.”

Referenced Topical Reports

- KP-TR-003-NP-A, Revision 1, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor”
- KP-TR-017-P, Revision 1, “KP-FHR Core Design and Analysis Methodology” (technical report)
- KP-TR-018-P, Revision 1, “Postulated Event Analysis Methodology” (technical report)

Staff Evaluation – Analytical Methods

- Core design methodology includes Serpent 2 and STAR-CCM+ (discussed in Ch. 4.5)
  - Serpent 2 is a 3D Monte Carlo code that models neutrons and photons
  - STAR-CCM+ is a computational fluid dynamics (CFD) code
- Transient analysis methodology includes KP-SAM and KP-BISON codes (discussed in Ch. 13.1.2)
  - KP-SAM is a modification of SAM. KP-SAM is an accident analysis code
  - KP-BISON is a modification of BISON. KP-BISON is a fuel performance code
- Codes have not been reviewed for verification and validation (V&V)
- The NRC is not approving the use of these computer codes

Staff Evaluation – Analytical Methods

- Hermes models account for the following heat transfer modes:
  - Pebble-to-pebble convective heat transfer
  - Pebble radiative heat transfer
  - Pebble-to-pebble heat transfer via pebble contact conduction
  - Pebble-to-pebble heat transfer via conduction through the coolant
  - Conductive, convective, and radiative heat transfer to reflector
- Staff reviewed thermal conductivity equations for pebble-to-pebble and pebble-to-coolant heat transfer
- Hermes models use the same pebble bed pressure drop correlation
  - Staff reviewed the pebble bed pressure drop correlation
- The NRC is not approving the use of these equations and correlations
- The NRC is not approving the references used to make the findings

Technical Evaluation Conclusions

- PDC 10, "Reactor design"
  - Hermes thermal-hydraulic design provides adequate heat removal
- PDC 12, "Suppression of reactor power oscillations"
  - Supporting analyses for power oscillations will be provided later
- PDC 34, "Residual heat removal"
  - PHTS removes heat during normal operation
  - Downcomer and fluidic diodes and DHRS enable heat removal during postulated events
- PDC 35, "Passive residual heat removal"
  - Downcomer and fluidic diodes and DHRS enable heat removal during postulated events
- Information provided is consistent with PDC 10, 12, 34, and 35 and with acceptance criteria of NUREG-1537
- Analytical methods for thermal-hydraulic design are acceptable at this stage of the design

T-H Design Regulatory Findings

- The NRC staff concludes that the preliminary design provided in PSAR Section 4.6 is consistent with the applicable PDCs and acceptance criteria in NUREG-1537.

- The NRC staff finds that the information in Hermes PSAR Section 4.6 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40, and further information can be reasonably left for the OL application

Questions?

NRC Staff Review for PSAR Section 4.7 Reactor Vessel Support System

Briefing for the Advisory Committee on Reactor Safeguards

March 24, 2023

By the Division of Advanced Reactors and Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Agenda

- Overview of PSAR Section 4.7 “Reactor Vessel Support System”
- Regulatory basis and PDCs
- Referenced topical reports
- Staff technical evaluation
- Technical conclusions
- Regulatory Findings

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Overview of 4.7 Reactor Vessel Support System

- The RVSS provides the structural support for the reactor vessel and the vessel internals.
- The RVSS supports the full weight of the vessel, fuel, coolant, vessel internals, and the head-mounted components.
- The RVSS is designed to transmit pressure, seismic, and thermal loads to the cavity structures and address thermal expansion during initial heat-up and postulated events.

Regulatory Basis

- 10 CFR 50.34(a) “Preliminary safety analysis report”
- 10 CFR 50.35 “Issuance of construction permits”
- 10 CFR 50.40, “Common standards”

- The following PDC (KP-TR-003-NP-A):
  - PDC 2 – “Design bases for protection against natural phenomena”
  - PDC 4 – “Environmental and dynamic effects design bases”
  - PDC 74 – “Reactor vessel and reactor system structural design basis”

- **Guidance:** NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors”

Staff Evaluation – PDC 2

PDC 2, “Design bases for protection against natural phenomena.”

- PSAR Table 4.7-1, “Load Combinations for the Reactor Vessel Support System,” provides the load combinations for the RVSS, including seismic loads due to design basis earthquake events
  - Will be used to demonstrate that the final design will meet the allowable stress limits specified in ASME BPVC Section III, Division 5
- Based on the acceptable load combination methodology, the staff finds that the preliminary information on the RVSS design is consistent with PDC 2

Staff Evaluation – PDC 4

PDC 4, "Environmental and dynamic effects design bases."

- Design features to address environmental and dynamic effects on the RVSS:
  - Pipe whip hazards are precluded by design due to a lack of pressurized piping.
  - Discharging fluids are addressed by catch basins with a leak detection system.
  - RVSS is designed to address temperature and mechanical loading cycles in order to prevent damage from creep-fatigue and to allow for thermal expansion of the reactor during startup and operation.
- Based on the design features to manage pipe whip hazards, discharging fluids, and loading cycles, the staff finds that the preliminary information on the RVSS is consistent with PDC 4 and the relevant NUREG-1537 criteria.

Staff Evaluation – PDC 74

- PDC 74 – “Reactor vessel and reactor system structural design basis”

- RVSS design helps ensure sufficient heat removal from the reactor vessel, while also providing structural support for the reactor vessel under static and dynamic loadings (e.g., seismic).

- The heat is removed via the reactor thermal management system (RTMS), actively during normal operation and passively during postulated events.

- Based on the design for sufficient heat removal and structural support, the staff finds that the preliminary information on the RVSS design is consistent with PDC 74.

Section 4.7 Regulatory Findings

- The NRC staff finds the preliminary design information is consistent with PDC 2, 4, and 74 and the applicable criteria in NUREG-1537.

- The staff concludes information in Hermes PSAR Section 4.7 is sufficient for the issuance of a CP in accordance with 10 CFR 50.35 and further information can be reasonably left for the OL application

Questions?

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NRC Staff Review for PSAR Section 6.2 Functional Containment

Briefing for the Advisory Committee on Reactor Safeguards

March 24, 2023

By the Division of Advanced Reactors and Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Agenda

- Overview of PSAR Section 6.2, "Functional Containment"
- Regulatory Basis
- Staff Technical Evaluation
- Technical Conclusions
- Regulatory Findings

Overview of PSAR Section 6.2

- Kairos Hermes uses a functional containment to limit potential release of radioactive material
- Functional containment includes physical barriers, operating conditions, coolant design, and fuel form
  - TRISO fuel retains radionuclides, layers form barriers, pebble provides physical protection
  - Large margin to TRISO fuel design temperature expected
  - Flibe credited for retaining radionuclides
  - Near-atmospheric primary system pressure
- PSAR Section 6.2 describes the strategy of using a functional containment; Chapter 13 intends to demonstrate its acceptability

Regulatory Basis

- 10 CFR Section 50.34(a), “Preliminary safety analysis report”
- 10 CFR 50.35, “Issuance of construction permits”
- 10 CFR 50.40, “Common standards”
- SECY-18-0096, “Functional Containment Performance Criteria for Non-Light-Water-Reactors” (ML18114A546) and its approval in SRM-SECY-18-0096, “Staff Requirements – SECY-18-0096 - Functional Containment Performance Criteria for Non-Light-Water-Reactors” (ML18338A502)

Staff Technical Evaluation: Review Process

- The NRC staff used relevant parts of NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” as guidance
- The NRC staff ensured the functional containment approach was consistent with SECY-18-0096
  - One difference: Maximum hypothetical accident (MHA), per NUREG-1537, instead of event categories
- The NRC staff ensured that the staff evaluated each individual component/feature of the functional containment as well as its integral performance
  - PSAR Section 4.2.1, “Reactor Fuel”
  - PSAR Section 4.3, “Reactor Vessel System”
  - PSAR Section 5.1, “Primary Heat Transport System”
  - PSAR Sections 13.1.1 and 13.2.1, both titled “Maximum Hypothetical Accident”

Staff Technical Evaluation: Conclusions

- PSAR Chapter 13 demonstrates the need for functional containment as an engineered safety feature
- Preliminary MHA analysis suggests radiological consequences are within 10 CFR Part 100 criteria; staff will confirm as part of operating license (OL) application
- Functional containment has no adverse impacts to normal operations or safe shutdown
- Safety related instrumentation to monitor functional containment components
- Technical specifications will confirm continued operability and are important to ensure that actual dose consequences are bounded by the MHA analysis

Regulatory Findings

- The NRC staff finds:
  - the preliminary design information is consistent with the applicable criteria in NUREG-1537
  - The functional containment approach is consistent with SECY-18-0096 and SRM-SECY-18-0096

- The staff concludes that the information in Hermes PSAR Section 6.2 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40, and further information can be reasonably left for the OL application.

Questions?

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NRC Staff Review for PSAR Section 6.3 Decay Heat Removal System

Briefing for the Advisory Committee on Reactor Safeguards

March 24, 2023

By the Division of Advanced Reactors and Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Agenda

- Overview of PSAR Section 6.3, “Decay Heat Removal System”
- Regulatory Basis
- Staff Technical Evaluation
- Technical Conclusions
- Regulatory Findings

Overview of PSAR Section 6.3

- The decay heat removal system (DHRS) is the safety-grade system credited to maintain the reactor vessel temperature within acceptable limits for SS-316
- DHRS is designed to provide passive cooling for up to seven days following a postulated event without electrical power or operator action
- Main components: water storage tanks, steam separators, annular thermosyphons
- Heat from the reactor vessel is transferred via radiation and convection to water in the annular thermosyphons, where water boils off
- DHRS is placed in service at a “threshold power” where passive radiative cooling alone is not adequate
  - During this transition phase, water is introduced to the guide tube and evaporator

Regulatory Basis

- 10 CFR Section 50.34(a), “Preliminary safety analysis report”
- 10 CFR 50.35, “Issuance of construction permits”
- 10 CFR 50.40, “Common standards”
- The following NRC staff-approved PDC from KP-TR-003-NP-A, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor”:
  - PDC 1, “Quality standards and records”
  - PDC 2, “Design bases for protection against natural phenomena”
  - PDC 3, “Fire protection”
  - PDC 4, “Environmental and dynamic effects design bases”
  - PDC 10, “Reactor design”
  - PDC 34, “Residual Heat Removal”
  - PDC 35, “Passive residual heat removal”
  - PDC 36, “Inspection of the passive residual heat removal system”
  - PDC 37, “Testing of the passive residual heat removal system”

Staff Technical Evaluation: Review Process

- The NRC staff used relevant parts of NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” as guidance
- The NRC staff reviewed the DHRS against the relevant PDCs
- The NRC staff reviewed the preliminary system design to identify potential system failure modes
- The NRC staff audited the Kairos detailed system description and DHRS heat performance calculation that determined the level of system performance necessary to maintain the vessel below the SS-316 limit
- The NRC staff performed independent calculations to ensure water tank volumes are sufficient for 7 days of cooling

Staff Technical Evaluation: Design-Specific Aspects

- The NRC staff ensured Kairos testing plans address several potential phenomena and failure modes, such as:
  - Potential flow and heat removal instability issues during the transition and in-service phases
  - Potential dynamic loads on the structure and components due to transition phase thermal shock and in-service evaporator boiling
  - Potential for corrosion and fouling in the evaporator tube
- The NRC staff noted that the DHRS design must accommodate the highest heat loads for vessel integrity and the lowest in-service heat loads to prevent freezing
  - Final determination on the adequacy of the DHRS to meet these competing design requirements will be made based on the final design presented in the OL application

Staff Technical Evaluation: PDCs

- PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to appropriate quality standards and identification of generally recognized codes and standards used
  - DHRS will be designed to American Society of Mechanical Engineers, American Society of Civil Engineers, and American Concrete Institute standards
  - Component design to these standards is evaluated in Chapters 3 and 4 of the NRC staff safety evaluation report (SER)
  - Quality assurance program is evaluated in Chapter 12 of the NRC staff SER
- PDC 2 requires protection against natural phenomena
  - DHRS located in safety related portion of reactor building except for steam vent lines
  - Failure of steam vent lines will not impede safety function
  - Seismic methodologies are evaluated in Sections 3.4 and 3.5 of the NRC staff SER

Staff Technical Evaluation: PDCs

- PDC 3 requires protection against fires
  - DHRS designed with low combustible materials and physical separation
  - Fire protection program is evaluated in Section 9.4 of the NRC staff SER
- PDC 4 requires protection against environmental and dynamic effects
  - DHRS will be designed to withstand normal operating conditions and to protect against dynamic environmental effects
- PDCs 10, 34, and 35 address adequate heat removal under normal operation and postulated events
  - DHRS is designed with redundancy to protect against single failures
  - Preliminary calculations in PSAR suggest that DHRS removes heat adequately
- PDCs 36 and 37 address inspection and functional testing of the DHRS
  - DHRS is designed for online monitoring and access to perform inspections
  - DHRS will be functionally tested during startup phase

Staff Technical Evaluation: Conclusions

- PSAR Chapter 13 demonstrates the need for DHRS as an engineered safety feature
- Preliminary PSAR analyses suggest that DHRS removes adequate amounts of decay heat and that radiological consequences are within 10 CFR Part 100 criteria; staff will confirm as part of OL application review
- DHRS has no adverse impacts to normal operations or safe shutdown
- Technical specifications and normal operation will confirm continued operability
- The Kairos DHRS test program will be key to finalizing the DHRS design

Regulatory Findings

- The NRC staff concludes that the preliminary design is consistent with the associated PDCs and acceptance criteria in NUREG-1537
- The NRC staff finds that the information in Hermes PSAR Section 6.3 is sufficient for the issuance of a construction permit in accordance with 10 CFR 50.35 and 50.40, and further information can be reasonably left for the OL application

Questions?

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NRC Staff Review for PSAR Chapter 9 Auxiliary Systems

Briefing for the Advisory Committee on Reactor Safeguards

March 24, 2023

By the Division of Advanced Reactors and
Non-Power Production and Utilization Facilities,
Office of Nuclear Reactor Regulation

Chapter 9, “Auxiliary Systems”

- Format
  - 9.1: Reactor coolant auxiliary systems (chemistry control, inert gas, tritium management, inventory management, thermal management)
  - 9.2: Heating, ventilation, and air conditioning
  - 9.3: Pebble handling and storage
  - 9.4: Fire protection
  - 9.5: Communication
  - 9.6: Possession and use of byproduct, source, and special nuclear material
  - 9.7: Plant water (service water, treated water, component cooling water, chilled water)
  - 9.8: Other (remote maintenance and inspection, spent fuel cooling, compressed air, cranes and rigging, auxiliary site services)

Overview of staff review

- Kairos applied for a CP and has not specifically requested approval of detailed design information
- Kairos provided a preliminary design description and a discussion of the relevant design bases (e.g., PDCs)
- NRC staff assessed whether the preliminary design information, including identification of relevant design bases, is sufficient to allow the staff to determine that
  - The information meets the relevant regulations for the issuance of a CP
  - The detailed design information can be left to the OL application

9.1.1 Chemistry Control System

- Used during normal operations to monitor coolant chemistry
- Ensures Flibe meets appropriate specifications
- Able to remove and replace reactor coolant to restore conformance to Flibe specifications
- Non-safety related system

Staff Evaluation of CCS

- PDC 2 requires protection against natural phenomena
  - Seismic methodologies are evaluated in Sections 3.4 and 3.5 of the NRC staff SER
- PDC 4 requires protection against environmental and dynamic effects
  - CCS will be designed to protect against dynamic environmental effects
- PDC 70 requires the CCS to monitor and correct reactor coolant chemistry
  - The CCS will measure coolant purity and can correct chemistry via the IMS if needed
  - Sampling locations, methods, frequencies and required actions will be available for staff review as part of OL application
- Proposed limits for reactor coolant chemistry are supported by CCS functions.
  - LiF to BeF2 ratio, circulating activity

9.1.2 Inert Gas System - Staff Evaluation

- PSAR Section 9.1.2 states the IGS is designed to:
  - Maintain inert environment, purge flow, remove impurities, tritium transport, and reactor coolant motive force
  - Assess purity, and process and store gases to ensure reactor and personnel safety
  - Ensure control and detection of leaks
- NRC staff has reasonable assurance criteria in NUREG-1537 can be met because:
  - IGS uses inert argon gas that is supplied to components with individually regulated temperatures, pressures, and flows
  - IGS contains storage tanks, equipment to measure and remove oxygen and moisture
  - IGS can be monitored for leaks and contains radiation monitors

Staff Evaluation of IGS (Cont'd)

- PDC 2 requires protection against natural phenomena
  - seismic methodologies are evaluated in Sections 3.4 and 3.5 of the NRC staff SER
- PDC 4 requires protection against environmental and dynamic effects
  - IGS will be designed to prevent pipe whip and argon gas is inert
- PDC 64 requires the IGS to monitor radioactive releases
  - IGS has radiation monitors and can be inspected to detect leaks
  - Proposed TS on circulating activity
- Proposed TS for argon volume and purity consistent with NUREG-1537 guidance that gas purity should be assessed.

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Reactor coolant auxiliary systems

- Non-safety-related systems
  - Tritium mitigation – monitor and remove tritium in vapor spaces
  - Inventory management – add and remove salt coolant
  - Reactor thermal management – preheat reactor, cool cavity
- Review criteria – NUREG-1537, Section 9.7, “Other Auxiliary Systems”
  - Not result in reactor accidents
  - Not prevent safe shutdown of the reactor
  - Not result in unacceptable radioactive releases or exposure

Reactor coolant auxiliary systems

- NRC staff review
  - Asked audit questions to enhance staff's understanding of PSAR
  - Checked to ensure PDCs listed in PSAR are appropriate to meet the review criteria
  - Checked to ensure PDCs listed in PSAR are appropriate for each facility-unique system
    - E.g., Thermal management system lists PDC 71, "Reactor Coolant Heating Systems"

Fuel handling and storage

- Pebble handling and storage system – circulate pebbles through core, load spent fuel in storage canister, transfer canister to pool
- Review criteria – NUREG-1537, “Handling and storage of spent fuel”
  - All special nuclear material accounted for
  - Fuel meets procurement specifications
  - Criticality prevented
  - Fuel-handling tools and procedures designed to avoid damaging fuel
  - Methods to assess fuel radioactivity and potential exposure rates adequate to avoid personnel overexposure
  - Shielding methods ensure doses below occupational exposure limits and ALARA

Fuel handling and storage

- NRC staff review
  - Asked audit questions to enhance staff's understanding of PSAR
  - Checked to ensure PDCs listed in PSAR are appropriate to meet the review criteria
    - E.g., PDC 62, "Prevention of criticality in fuel storage and handling"

Staff conclusions

- The preliminary design of the auxiliary systems is sufficient and meets the applicable regulatory requirements and guidance for the issuance of a construction permit in accordance with 10 CFR 50.35.
- Further technical or design information required to complete the safety analysis can be left for later consideration in the FSAR.