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

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
(ACRS)
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KAIROS POWER LICENSING SUBCOMMITTEE
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TUESDAY
APRIL 18, 2023
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The Subcommittee met via hybrid in-person and Video Teleconference, at 1:00 p.m. EDT, David Petti, Chairman, presiding.

COMMITTEE MEMBERS:

DAVID PETTI, Chair
RONALD G. BALLINGER, Member
CHARLES H. BROWN, JR., Member
VICKI BIER, Member
VESNA DIMITRIJEVIC, Member
GREGORY HALNON, Member
WALT KIRCHNER, Member
JOSE MARCH-LEUBA, Member
JOY L. REMPE, Member
MATTHEW SUNSERI, Member

ACRS CONSULTANT:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

LARRY BURKHART

ALSO PRESENT:

ODUNAYO "AYO" AYEGBUSI, NRR

BENJAMIN BEASLEY, NRR

ANDREW BIELEN, RES

MATTHEW DENMAN, Kairos Power

KIERAN DOLAN, Kairos Power

TIMOTHY DRZEWIECKI, Kairos Power

JORDAN HAGAMAN, Kairos Power

MICHELLE HART, NRR

BRANDON HAUGH, Kairos Power

EDWARD HELVENSTON, NRR

MATTHEW HISER, NRR

DREW PEEBLES, Kairos Power

JEFFREY SCHMIDT, NRR

KENNETH CHARLES WAGNER, Kairos Power

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

1:00 p.m.

CHAIR PETTI: Okay. This 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, Joy Rempe, Matt Sunseri, Ron Ballinger, Walt Kirchner, Vesna Dimitrijevic, Vicki Bier, and Greg Halnon. Our consultants, Dennis Bley and Steve Schultz, are also present. Weidong Wang of the ACRS staff is the Designated Federal Official of this meeting.

During today's meeting, the subcommittee will continue its review of the staff safety evaluation on the Kairos Power 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.

A 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 552(b)(c)(4).

Attendance at the meeting that deals with such information will be limited to the NRC staff and its consultants, Kairos Power, and those individuals and organizations who have entered in 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 have 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 kept and will be made available.

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

We'll now proceed with the meeting. Ed, do you want to say something to kick us off?

MR. HELVENSTON: I have no introductory remarks for the staff, so I think we'll turn it over to Kairos for the presentation on Section 12.9, Quality Assurance.

MR. HAGAMAN: Thank you. Good afternoon. My name is Jordan Hagaman. I'm the Director of Reliability Engineering and Quality Assurance and Kairos Power. Today, we're talking about Section 12.9 of the PSAR. For a broader context, Chapter 12, in general, describes all the plans for conduct of operations at Hermes. This includes facility operating, emergency planning, security plan, QA plan, operator training, requalifications, startup, and environmental reports.

CHAIR PETTI: We don't see any slides.

MR. HAGAMAN: Okay. Let me pause there and see.

CHAIR PETTI: This is interesting. Those that have computers in the room see them, but we're not getting them on our screens. So let's pause a minute.

PARTICIPANT: I think there may have been two different schedulers. There was one that said, it said placeholder or something. It's possible that we're in the wrong --

CHAIR PETTI: Well, except that this is where the court reporter is and this is where -- let's see. Any of the ACRS virtual members online? Matt, Vesna, Walt?

MEMBER SUNSERI: Yes, this is Matt. I see the slides.

MEMBER DIMITRIJEVIC: Yes.

CHAIR PETTI: Okay. So I think we're in the right place.

MEMBER KIRCHNER: Yes, I do, too. This is Walt.

(Long pause.)

MR. HAGAMAN: All right. Once again, my name is Jordan Hagaman. And the main thing I wanted to point out at the title slide is we're looking at

Section 12.9, which is just one small part of Chapter 12, which describes all of the conduct of operations for the Hermes plant.

So with that, we can jump to the next slide. 10 CFR 50.34 requires construction permit applicants to provide a QA program description to be used to design, build, and operate the structure systems and components for the reactor. We started with NUREG 1537, which pointed us to guidance in Reg Guide 2.5 and ANS 15.8, which was used to develop the format and content of the quality assurance program description for the Hermes non-power reactor. This is provided in full as an appendix to Chapter 12.

On the applicability of this QA standard, ANS 15.8 describes that the type of QA program appropriate to a research and test reactor is different than the type of QA program applied to commercial power reactors. The front matter of the standard describes the characteristics that are different between non-power and commercial power reactors that affect the type of QA program recommended. The key of these characteristics could be summarized as the relative simplicity of the safety case for research and test reactors, which is fundamentally different than the safety cases for larger commercial power reactors. The safety characteristic of research and test reactors could also be applied to Hermes.

We'll discuss later today in the Chapter 13 presentation about the preliminary safety analysis prepared for Hermes that shows very large margins to Part 100 dose consequence limits. This is the key metric for a simplified safety case, helping us to establish that the Hermes safety profile is similar to that of other research and test reactors.

We can go the next slide. The Hermes quality assurance program description applies to design phase, construction phase, and operations phase activities affecting quality for safety-related structures, systems, and components. I'd like to briefly expand on that to help describe the applicability of the program. We've discussed the Hermes definition of safety related in previous subcommittee meetings. That definition of safety related is repeated in the Hermes QAPD for consistency. To summarize, it includes all SSCs that are responsible for at least one of three things: the first one being SSCs responsible for the integrity of the vessel, maintaining coolant above the core; the second being SSCs responsible for reactivity shutdown capability; and the third is SSCs that provide capability to prevent or mitigate accident consequences beyond Part 100 limits.

As far as program applicability is concerned, there's a table in Chapter 3 of the PSAR. That's Table 3.6-1. This table lists all of the SSCs for Hermes and notes both safety classification and quality program applicability. You'll note that all SSCs designated as safety related are also listed as quality related. Therefore, the requirements of the Hermes QA program apply to those SSCs.

Examples of the safety-related SSCs are the reactor vessel, the reactivity shutdown elements, the decay heat removal system, the reactor protection system. Quality-affecting activities associated with those SSCs include the final design, fabrication, construction and testing.

We can go to the next slide, please. All right. As mentioned in the previous slide, the Hermes quality assurance program description describes requirements for design, construction, and operations phase activities affecting quality. However, at the CP stage, only the design and construction portions of the QAPD were subject to review. As a result, the requirements for facility operations do not appear on this slide, but we do look forward to discussing those requirements during the review for the operating license.

Also not listed here is the 19th requirement in the design and construction section of ANS 15.8. Requirement 19 is custom for research and test reactors. That's for experimental equipment. The Hermes demonstration reactor is not being designed or licensed for experiments. Rather, the project mission is to demonstrate the construction and operation of a Kairos FHR and to demonstrate delivery of low-cost nuclear heat. Without formal defined experiments, Requirement 19 for experimental equipment does not apply. What does apply are the traditional 18 QA criteria that we're familiar with. The requirements described in ANS 15.8 are, more or less, directly accepted into the Hermes quality assurance program with only editorial changes.

And with that, that's the end of my prepared remarks.

MEMBER HALNON: Hey, Jordan, this is Greg Halnon. Did I hear you right that there's only two of the criteria that are in play right now, and it's the design and what was the other one?

MR. HAGAMAN: Design and construction were the ones that are subject to review during the construction permit stage.

MEMBER HALNON: Okay. So I understand your point on, you know, they'll be more operationally phased. I'm interested in the corrective action portion. How, if that's not in -- do you have a corrective action program now that will just carry over to the operation phase, it's just not subject to review right now, or are you waiting to put that in place later on?

MR. HAGAMAN: So the third from the bottom on the right-hand, the corrective action program, Requirement 16, is part of the design and construction phase.

MEMBER HALNON: Okay. Thanks.

CHAIR PETTI: Members, any other questions?

MEMBER BALLINGER: Yes, I have a --

CHAIR PETTI: Go ahead.

MEMBER BALLINGER: -- I guess it's a theoretical question. So it's not designed for experiments. So you build this thing and you start operating it, and you find out that something doesn't work and that not work would translate into the FHR. Are you saying that you cannot, because of the restrictions, you cannot do an experiment or what would be called an experiment with this plant to solve a problem which you've discovered that will translate into the FHR?

MR. HAGAMAN: So we would expect anything that gets implemented in terms of modifications for the Hermes plant to be subject to the same reasonable assurance that it's going to perform a safety function as any of the originally-designed SSCs. So we will have that reasonable assurance before we put those SSCs into service, so they wouldn't be considered an experiment. They'll be just the same as any other SSC that was part of the original design.

CHAIR PETTI: But I think Ron's question was a little different. Let's say you find, you know, something doesn't go as planned, not just related necessarily to SSCs, but something where, in order to fix it, you might have to go outside your tech spec and have to change the tech spec. There's a process for that, I would think, right, so that you could do that?

MR. HAGAMAN: That should fall under our normal 50.59 process.

MEMBER BALLINGER: Yes, okay.

CHAIR PETTI: Any other questions, members?

MEMBER REMPE: The staff talk about it,

but, if you explore Ron's question, I thought we kind of discussed this a while back. If the reactivity coefficients are as anticipated or you want to try and better understand some instabilities you see, there are tech specs and you are going to have to try and get more data. It seems like that, you know, with an operating plant, the staff would be cognizant of that ahead of time. The applicant knows they have to discuss this with the staff, the staff would say, yes, okay, you're going to be doing some sort of test. It's the whole reactor is sort of an experiment, and they have to communicate it to the staff, and the staff would have some process in place ahead of time before the licensee would be able to do that test with the entire reactor. And I thought the staff had told us at that time in whatever chapter it was that, yes, they need to do that and that will be clear.

MR. HELVENSTON: Yes, I think when we talk about there not being experiments, you know, we sort of mean in the traditional sense where, you know, they're not necessarily doing some of the, you know, sample irradiations, radiography, isotope production, things like that that you'd associate with a traditional operating non-power reactor. But in a sense, like you said, it really is the reactor itself would be considered somewhat of an experiment. And we expect Kairos, Your Honor, in the operating license phase to have startup plans and sort of, you know, procedures in place to look at, as they're starting up, in a phase approached and taking observations and learning as they go, and, you know, that could still inform the future operations of the facility.

And, certainly, you know, the NRC has, you know, the regulations have processes in place, like the 50.59, the license amendment process, you know, if there needs to be some change to how the reactor is operated or some system based on the operational experience that's been collected up to that point.

MEMBER REMPE: Thank you.

CHAIR PETTI: Okay. Then did the staff have any slides on QA? Thank you. Go ahead.

MR. HELVENSTON: Are you sharing the slides, Ben? You can go ahead to the next slide.

So I'll just start off like we did on, I think, the previous meeting, just go into the agenda and a couple of the highlighted level items that apply to all the sections we're going to be presenting over the next couple of days, you know, to avoid having to do this at the beginning of each section again. So I'll just briefly go over, we'll start out with a presentation on PSAR Section 12.9 on quality assurance. Also, later this afternoon, we'll provide a presentation on the sections of the PSAR of the Chapter 13. They're specific to the maximum hypothetical accident. And then I believe tomorrow morning we'll follow that up with a discussion of the remaining sections of Chapter 13 on the postulated bounded events.

In terms of the agenda for each chapter of the staff's presentation, that will be pretty similar. We'll start with an overview of the chapter and the relevant PDCs, if there are any; any topical reports; what we did for our technical evaluation; and then the staff's findings and conclusions.

Next slide. So in terms of the reg basis that we looked at in our review of these chapters, the three regulations that are in common for every section we looked at is 50.34(a), 50.35, and 50.40, as well as the guidance in NUREG 1537, Part 2, which provides the review plan and the acceptance criteria for the application. In some of the subsequent presentations, there may be some additional regulations and guidance that are applicable to that specific section that we'll go into detail on the following presentations.

So with that, I'll turn it over to Ayo who will present on the NRC staff's review of PSAR Section 12.9 on quality assurance.

MR. AYEGBUSI: Thanks, Ed. So good afternoon. My name is Ayo Ayegbusi. Can you hear me? All right. Like I said, good afternoon. My name is -- is this better? All right. My name is Ayo Ayegbusi, and I am a reactor operations engineer in the Quality Assurance and Vendor Branch in NRR. My presentation today will discuss the staff's review of the quality assurance section in the Kairos Hermes PSAR.

Next slide, please. All right. So in Section 12.9 of the PSAR, Kairos states that its quality assurance program is based on Reg Guide 2.5 which endorses ANS 15.8, which is the quality assurance program requirements for research reactors. The Kairos Hermes QAPD is described in Appendix B of PSAR Chapter 12. So that's just background information, some of which Kairos has covered.

Next slide, please. In addition to the regulations and guidance mentioned earlier during the common regulatory basis that Ed covered, the staff specifically reviewed the Kairos Hermes QAPD against the ANS 15.8 standard.

Next slide, please. So for the staff's

evaluation, the staff evaluated Sections 1 and 2 of the QAPD because those two sections directly apply to the construction permit application. As Kairos mentioned, those sections cover design and construction.

The staff's evaluation found that Kairos followed the ANS 15.8 standards closely. My many slides will cover areas where Kairos deviated from the ANS 15.8 standard. However, the staff did not evaluate Section 3 of the QAPD because it covers facility operations, which, at this point, is not relevant to issuing a construction permit.

Next slide, please. The first deviation from the ANS 15.8 standard is that Kairos proposed an alternate definition for safety related to match what is used in PSAR Chapter 3. The staff found this proposal acceptable because it's consistent with the Hermes design and the safety related definition in the ANS 15.8 standard. At this point, my understanding is that ACRS has been given a draft copy of our safety evaluation. That does not include our evaluation in what we found here, but we will be revising that safety evaluation to discuss our findings as far as it relates to the safety related definition that Kairos proposed.

The next deviation from ANS 15.8 standard is that the QAPD did not include a section for experimental equipment. Kairos already covered that. Again, the staff found this acceptable because the PSAR states that no experiments will be carried out, and I'm paraphrasing that.

The next deviation from the ANS 15.8 standard is that the QAPD did not include Section 4 and 5 from the standard, namely applicability to existing facilities and decommissioning respectively. In this case, the staff found this acceptable because the QAPD will not utilize an existing facility, and decommissioning plans are not required for the construction permit application. And I --

MEMBER HALNON: This is Greg, just real quick. You paraphrased to say that they're not going to do experiments. Is the demarcation between experiment and test clear enough such that we're not going to be arguing on whether it's an experiment or a test? Because it's like a 50.59 experiment test and modification, so is that clear enough in the regulation for them to be able to ascertain that no experiments will be done?

MR. AYEGBUSI: So like I mentioned earlier, because I hear you mentioned regulation, our review was based on the ANS 15.8 standard, which is what we've endorsed, right. Section 2.19 has to do with, I forget the title, but it has to do with experiments, right. I think it's experimental equipment, equipment for experiments. And so that's focused on experiments, right. It doesn't address testing.

So to your question for testing, I would have to defer to Ed. What he said earlier is they would have to address it --

MEMBER HALNON: It's not a real fine point. I'm just curious because, at least in the operating reactor world, in light water, we always had that struggle internally. When we did test procedures, someone said is this experimental or not, and we never really found a good demarcation of where that line was between a test and experiment. Now, it may be in the test reactor world it's much more clear, and that's what I was kind of getting to, if that's more clear in the test reactor world, or the research reactor, I'm sorry.

MR. AYEGBUSI: Honestly, I would have to defer to the other technical staff because this section really focuses on quality assurance, so, in essence, the quality of the activities of the design and construction of the plant so --

MEMBER HALNON: Okay. Well, you said they were clearly within 2.9, and I'm satisfied with that. But maybe at another time I'll have that philosophical discussion. Maybe there's some hard information somewhere we can get to.

MR. AYEGBUSI: Okay. Next slide, please. I already spoke to this slide, so I'm going to go on to the next slide, please.

So the staff's safety evaluation recommended that the construction permit should include a condition for the quality assurance program. The condition requires that the QA program is implemented, as described in the PSAR, and any changes that reduce the commitments in the QAPD are submitted to NRC for approval prior to implementation.

Next slide, please. So in conclusion, the staff found the preliminary design information to be consistent with the applicable criteria in NUREG 1537. The staff concluded that the information in Section 12.9 and Appendix 12(b) of the PSAR is sufficient for the issuance of a construction permit. Lastly, the staff concluded that reviews related to the conduct of operations and decommissioning can be left at the operating license application phase.

Next slide, please. So that concludes my presentation. Thank you. And are there any questions?

MR. SCHULTZ: I presume that Kairos has accepted the recommended change that you indicated in terms of changes to the QA program that can be made without NRC approval and then submitted 90 days prior or subsequent? On the previous slide it was described.

MR. HELVENSTON: Yes. We have a proposed recommended permit condition that's described on that slide, but that is something that we would likely verify with Kairos before that's finalized to make sure they understand and are in agreement with that condition.

MR. SCHULTZ: That sounds like a good idea. Thank you.

CHAIR PETTI: Other comments, members? Okay. Thank you. With that, we can go to the Chapter 12 memo. The Chapter 12 memo does not have a section explicitly on Section 12.9, so I think, in expediency, it's probably not worth, we've already seen the Chapter 12 memo in March, so I think we can just keep the schedule moving and move on to Chapter 13.

MEMBER SUNSERI: Dave, this is Matt. You're correct. I mean, we did address the QA program in that memo. We kind of got ahead a little bit, so it's already been discussed and incorporated into the memo.

CHAIR PETTI: Thank you. So let's move on to Chapter 13 then. Kairos.

MR. DENMAN: Hello. My name is Dr. Matthew Denman. I'm a reliability engineer at Kairos Power, and it's my pleasure today to talk to you about the Hermes Chapter 13 PSAR accident analysis.

Next slide. In 10 CFR 50.34(a)(4), it requires a preliminary safety analysis to assess the risk to public health and safety from the operation of a facility and determination of the margins to safety. In order to demonstrate compliance with 10 CFR 100.11 dose reference values, a maximum hypothetical accident was developed that bounds the postulated events, and this is analyzed for dose consequences by challenging the performance of our functional containment. The Hermes MHA approach is consistent with the guidance in NUREG 1537. It's not a physical accident. It is hypothetical in nature. It includes conservatisms that maximize the source term and the release off-site, and it includes a postulated release of radioactive material.

To ensure that postulated events are indeed bound by the MHA, we developed a list of postulated events that is comprehensive to ensure that any event with potential significant radiological consequence will be considered. Initiating events and scenarios are grouped so that limiting cases for each group can be qualitatively described in the construction permit application. Quantitative results will be included at OL. Acceptance criteria are provided for the important figures of merit in each postulated event group to ensure that the potential consequences of that event group remain bound by the MHA as the design progresses. Prevention of event initiators are also justified in the PSAR.

Next slide.

MR. SCHULTZ: Matt, before you leave that slide, this is Steve Schultz. You've indicated in that last group of bullets that, when you go to the operating license application, you're going to provide the quantitative results. Is that going to be group by group or by accident by accident? How are you going to present those quantitative results?

MR. DENMAN: Thank you very much for that question. The OL will present the results group by group, but we will have internal analysis to justify that our grouping or that the presented results is bounding of the group.

MR. SCHULTZ: Good. Thank you.

MR. DENMAN: Okay. So just another slide to kind of conceptualize this relationship between the dose limits, the maximum hypothetical accident, and our postulated events, the MHA is constructed to be extremely conservative and non-physical to overestimate the potential off-site dose consequences, ensure that we have sufficient margin to safety, and ensure that reasonable design constraints will result in a bounded postulated event.

If you look over at the qualitative figure on the right, you'll see that we've got our 100.11(a)(1) and (2) dose reference values. That's a mouthful. There's going to be a sufficient margin between those reference values and where our MHA dose is going to occur. Because of the hypothetical and conservative assumptions that go into the MHA that will not be included in the postulated events, you're going to have a standoff of additional dose, and then you'll have a range of doses or calculated doses where the potential postulated events will arise, and these will be due to our traditional design basis conservatisms that go into both the thermal fluid calculations and our mechanistic dose or source term methodology. At the PSAR stage, only the MHA dose is quantitatively evaluated, and this is the only event that is needed to ensure that sufficient margin exists to the 100.11 dose reference values.

MEMBER MARCH-LEUBA: Hi, this is Jose. You say on the PSAR stage. Is there any other stage where the dose would be for other postulated events?

MR. DENMAN: Thank you very much for that. As was mentioned on the previous slide, we are proposing a series of figures of merit which will, assuming that we -- sorry, not assuming. We will demonstrate that those figures of merit meet certain acceptance criteria and that, by going to the figures of merit NEPA acceptance criteria, that will map to a dose less than the MHA.

MEMBER MARCH-LEUBA: So only the MHA will be evaluated. The rest will have to do with figures of merit?

MR. DENMAN: That is what we described.

MEMBER MARCH-LEUBA: Thanks.

MR. DENMAN: Okay. Next slide. So the maximum hypothetical accident. I've got a couple of slides on the overall narrative here. A key feature of the maximum hypothetical accident is this time-temperature curve or curves. There's one for the fuel and one for the coolant. What should be noted here is that explicit system performance is not modeled. In fact, the boxy nature of the time-temperature curves are slightly intended to demonstrate that we're not mechanistically modeling our system performance and our temperature history. Instead, these temperature curves are designed to ensure that a bounding radionuclide release from our functional containment will occur, so it's not just the high temperatures but it's the extended and exaggerated time intervals over which we're at these high temperatures will ensure that the functional containment will be maximally stressed and off-site doses will be conservatively high.

MEMBER REMPE: Before you leave this slide, could I ask a couple of questions? I struggled on where to bring this up, but I think this temperature plot is the best place to bring this up.

When I look at your various scenarios or challenges and events you have, you have an event where you have air ingress into your primary system, and you note that the graphite oxidizes, as well as the carbon matrix. And I don't see anywhere in the PSAR or that topical report you generated or even in the staff SE about combustible gas generation that would occur when you oxidize the graphite CO with CO2. And I'm wondering, I guess you haven't done it yet because I'm guessing you're waiting, because you've created this maximum hypothetical accident that's going to bound all possible challenges, so I'm guessing you didn't use your codes to evaluate how much combustible gas got generated, and I don't think there's any system that I've seen in your description of what you're going to do with the combustible gas that gets generated and I'm not sure you know how much is. And I'm just thinking that somebody needs to think about combustible gas generation, and maybe it's a small amount, but anytime you get above 500 - 600 C, which this plot has, that could be a problem, especially when you get up to temperatures like 1,000 C or whatever, 850 or whatever.

And so, anyway, I'm just kind of thinking that somebody needs to think about combustible gas generation and if it could be an issue.

MR. HAUGH: Hi, Joy. This is Brandon Haugh, Director of Modeling Simulation. Great point, good question. We are considering that. It's, you know, it requires a lot of, I'm going to say, design fidelity to understand the predictability of that, but we are creating models and, if we deem that's a risk, we'll be able to understand how much is generated.

MEMBER REMPE: So you don't know, and you might have to add a system or you might want to try and have a primary system that can withstand the shock of an ignition, a combustion event or something. It just seems like somebody ought to do some scoping calculations early on before you pour concrete and start ordering components on this.

MR. HAUGH: It's great feedback. We have done that and we are doing that.

MEMBER REMPE: So how much gas do you get and where does it go, if you've already done that and you don't think it's a problem?

MR. HAUGH: Well, it's highly dependent on the chemistry and the temperatures in the system because it re-oxidizes back to be non-combustible depending on the situation. So it's very scenario-dependent on the amount of air ingress and the temperature time history. So there's a good amount to unpack there, and it's probably more than this discussion is needed, but it will be covered at the operating license application phase.

MEMBER REMPE: Okay. So, again, I'm just one member, but I strongly recommend that the memo can point this out and that our letter point this out because this is something that I don't see in Appendix A, and it's something I think people ought to make sure gets addressed. And I'll stop there. Thank you.

CHAIR PETTI: So just another point, this is, of course, a big deal in helium gas-cooled reactors, and you have to go way back, but my understanding is studies were done, I want to say by Brookhaven, the CO that's generated is usually on the lean side, so it's not combustible, at least that's what they found in HTGRs. So it's probably worth you guys trying to find that information, as well, and understand that chemistry, as you think about the chemistry.

MR. DENMAN: Thank you very much. One other point I just want to clarify is that the time-temperature curves you're seeing here are bounding temperatures for our fuel and our fuel covered by our Flibe, right. So pebbles that are suspended above the Flibe would, A, not expected to be at these temperatures and, B, would be handled separately from the MHA analysis.

MEMBER REMPE: Yes, but I don't have any curves to show me what those temperatures are, I don't have a risk assessment to show that the frequency of such events is very low. So, again, I need more information. Of course, you can go ahead and pour concrete, you can wait until the operating license; but I think it definitely is something that everybody needs to think about and have a good answer on.

MR. DENMAN: Understood. Thank you for that comment.

MEMBER BALLINGER: This is Ron Ballinger. You know, this curve puts you squarely at the upper limit on the stainless steel, and so you're into Division 5. But the best estimate for some of these things is considerably lower. So with this bounding calculation, you're definitely having to consider creep; is that right?

MR. DENMAN: Well, first off thank you very much for your question. I'll note two things. One, the MHA is designed to maximize release of radioactive material from our functional containment. It is not an accident that is designed to analyze stress on vessels or other components within the system. In fact, our commitment on our vessel temperature in the CPA is lower than the 816 ASME steel temperature limit. We are using this higher temperature as the stressor on our functional containment and then that delta between where the temperatures actually are going to wind up in our system and these evaluated temperatures will play into why our MHA will end up releasing more radionuclides than our postulated events will.

MEMBER BALLINGER: Okay. Thanks.

CHAIR PETTI: So I view it as sort of an artificial thing, right.

MEMBER BALLINGER: You could artificially fail the --

CHAIR PETTI: Well, right, in this hypothetical sense. But, yes, the few curves that are in the appendix of the technical report I think shows there's good margin there, and I actually noted that in our letters. Keep going.

MR. DENMAN: Okay. Thank you. So for our maximum hypothetical accident, we have radionuclides that are postulated to diffuse from TRISO particles. The distribution of TRISO particles included in the MHA account for both manufacturing defects and potential in-service failures prior to the transient occurring.

Pre-transient diffusion of radionuclides from kernels are hypothetically and conservatively not modeled to maximize the fuel inventory available for release during the MHA. Radionuclides are postulated to evaporate and de-gas from the Flibe, as driven by the conservative natural circulation boundary conditions. No hold up of any gases are credited within the Flibe portion of the functional containment.

Tritium is conservatively assessed to maximize both its initial inventory and its subsequent release. The initial inventory of Tritium is conservatively assessed and released. Tritium is conservatively postulated to desorb from in-vessel graphite as a function of temperature and instantaneously release from both steel and Flibe.

CHAIR PETTI: Matt.

MR. DENMAN: Yes, sir.

CHAIR PETTI: The question on the tritium. Did you include all the sources besides the Flibe? Did you look at lithium impurity in graphite and ternary fission sources? I'd like at the ternary fission, and I scaled it. I don't think it's an issue. I always am not sure on the lithium and graphite.

MR. DENMAN: I agree with you. The ternary fission is very insignificant and, in fact, that is part of our fuel inventory that subsequently would diffuse out through our grouping structures. The lithium impurities in the graphite is considered.

CHAIR PETTI: Okay. That's good. I mean, in the old days, there used to be a lot of lithium in graphite, and people got very worried about it. I don't think it's a problem today. I think they're just better quality graphite. But if you go and read the old literature, you can get a little confused that it's still a problem. I don't think it's as problem. I'm glad you confirmed it. Thanks.

MR. DENMAN: Okay. And then --

MR. SCHULTZ: Matt, this is Steve Schultz. It's not stated on this slide, but, in the documentation, with regard to the TRISO particles, the TRISO behavior during the accident, the release is from diffusion only, and then it would be a different release if the particles are failed before the accident. But the particles do not fail during the accident; is that correct?

MR. DENMAN: That is correct. The diffusion is an effective diffusion term, so it, you know, accounts for multiple different ways radionuclides can move through the system and just approximate it as a diffusivity. In-service failures are pre-transient. We do not expect there to be a statistically-significant fraction of in-transient failures, as will be shown in our postulated events.

Thus, the transient failures are not included in the MHA.

MR. SCHULTZ: Okay. And you demonstrate that statement regarding the events that are evaluated will demonstrate that there isn't going to be particle failure as a result in those events that we'll see evaluated at the operating license stage?

MR. DENMAN: Correct.

CHAIR PETTI: So, Steve, if you look at the database on TRISO, the failure rates under the accidents that go up to 1600 degrees is like 10 to the minus 5, and they're assuming 10 to the minus 3 order, so it's down in the --

MR. SCHULTZ: Good. Thank you, Matt.

MR. DENMAN: Thank you. Okay. So going through the methodology in a little bit more detail, the Hermes MHA uses a methodology or methodologies from the approved KP-FHR mechanistic source term methodology topical report, KP-TR-12-P-A. The concepts, the following concepts will directly leverage the topical report. This includes our radionuclide grouping and transport approaches for our TRISO fuel and our Flibe coolant mass transfer correlations for tritium into graphite reflectors and pebbles. That's part of the inventory calculation.

Gas face is not credited for confinement of radionuclides that release from the Flibe free surface and a two-hour hold up assumption for radionuclide transporting through the reactor building is modeled. Conservative unfiltered ground-level releases are modeled to maximize off-site doses. So all of these come directly from that topical report.

MR. SCHULTZ: Matt, Steve Schultz again. The ground-level release assumption, is that based upon the configuration that you expect from the facility? In other words, that's where you would expect to see the release? It's not apparent to me that that maximizes off-site doses at ground-level release versus an elevated release.

MR. DENMAN: Steve, thank you very much for the question. The ground-level release is not indicative of what we would expect a release to look like from the facility. However, as part of the topical report, we and the staff agreed that this was a suitably conservative approach.

MR. SCHULTZ: Okay. Maybe the staff will come in on that one in their presentation. Thank you.

MR. DENMAN: Okay. The following additional non-physical conditions provide additional hypothetical challenges to the functional containment beyond which is described in our mechanistic source topical report. The hypothetical time-temperature histories are applied to the transient. You've already seen a preview of that, and we'll go back and show you a little bit more in subsequent slides. This ensures that the MHA will bound the system temperatures from postulated event groups. The pre-transient diffusion of radionuclides from the fuel and the reactor core is negligible. This ensures that the maximum inventory is available for release at the initiation of the transient. A bounding vessel void fraction is assumed to facilitate the release of low volatile species in the vessel via our bubble burst release model. And additional conservatisms in tritium modeling are used to address limitations associated with the tritium modeling in graphite as described in our approved topical report.

CHAIR PETTI: What was that specific, the tritium that was adjusted, if you will?

MR. DENMAN: So we have a couple of things. May I table that to the next few slides --

CHAIR PETTI: Sure.

MR. DENMAN: -- and we'll talk a little bit more when we get to the inventory discussion, as well as the release discussion. I don't want to have to try to --

CHAIR PETTI: Yes, no problem.

MR. DENMAN: Thank you. Okay. So our basic approach for the MHA is kind of three stages. The first stage, we identify and account for all sources of material at risk and all barriers that that material is going to see as it releases through the system. We're going to evaluate release fractions for every combination of barrier radionuclide group and time interval associated with the MHA, and then we're going to use the RADTRAD and ARCON code to evaluate our dose consequences at the exclusionary boundary and the low population zone.

And then here we have kind of a graphical representation. All of our MAR and fuel kernel is first going to be held up in our TRISO fuel. Then it's going to propagate into the Flibe into the gas face. Circulating activity is going to start in the Flibe but then can evaporate or de-gas into the gas face. Our structural MAR is tritium and argon-41. These are both gases, so, once they release from the graphite, they bypass the Flibe and move directly into the gas face.

Diving a little bit deeper into our sources of material at risk, most of our material at risk in our system is contained within the TRISO fuel. The Serpent 2 code is used to evaluate fuel inventories for our reactor. Pre-transient depletion of radionuclides from the fuel is neglected in order to maximize the inventory of available material at risk. The circulating activity uses a bounding circulating activity distribution of radionuclides. This is expected to be controlled by technical specifications. And, importantly, the circulating activity, because this is a bounding value, is accommodating what we expect to see from nominal release of radionuclides from the TRISO fuel into the Flibe coolant. So any radionuclides that would have nominally left the TRISO fuel into the Flibe coolant during normal operations are effectively being double-counted here because that TRISO fuel assumes that there's no depletion of that radionuclides.

Next slide. So we also have our structural MAR. We'll focus on tritium first. The inventory conservatively bounds the operating lifetime at a full capacity factor with margin while accounting for differential uptake rates of our pebbles and reflector. The transfer from Flibe to structures, the tritium is assumed to be born in the Flibe but transferred to and absorbed into structures.

Primarily, this is the graphite that's going to uptake the tritium and store it up for release in the maximum hypothetical accident.

Transport speciation is conservatively assigned to tritium fluoride to maximize the tritium absorption into our system, e.g. our graphite. And then transfer from Flibe to structures is determined by max transfer coefficients from our predicted Flibe flow characteristics at steady state in our reactor.

When we talk about absorption within structures, the tritium absorbs solely as a function of mass transfer from the Flibe to structures, i.e. there's no diffusion resistance. If it can transfer in, it gets stored and locked up. And then retention of that tritium is modeled without any steady-state release mechanism, so this a perfect absorber of graphite. It just sucks in tritium due to mass transfer during the operation of the facility and then this should maximize the quantity of tritium that then would be available for release during the transfer.

CHAIR PETTI: So, Matt, just a question. In reality, there will be partitioning between the Flibe and the graphite, and the question is if, in fact, you made the graphite less sorb to, so the inventory was higher in the salt, doesn't it come out of the salt easier than it comes out of the graphite? So is that truly a conservative assumption to put it all in the graphite? Wouldn't it be more conservative to keep it in salt? Did you look at that sort of stuff?

MR. DENMAN: Thank you very much for the question. I would note that the quantity of tritium that we end up absorbing into the graphite are orders of magnitude higher than the quantity of tritium that is expected to be circulating through the salt. We did look at a lot of these sensitivities, and it was much more conservative, given these set of boundary conditions have as much tritium absorbed into the salt. Also, due to our -- sorry. Not in the salt, in the graphite.

Also, due to our highly-conservative release models, the tritium in the graphite gets released in a non-physical rapid rate. So even though, yes, the graphite is going to hold it a little bit more than the Flibe as we model the system in the MHA, it's not that much more.

CHAIR PETTI: Okay.

MR. DENMAN: And we can talk a little bit more about that as we get to the release models.

CHAIR PETTI: Okay. Thanks.

MR. DENMAN: Not a problem. And thank you for your question. Okay. So moving on -- no, no, argon-41.

MR. PEEBLES: Before we move on from that, we did have a correction to make. So, Kieran, are you online?

MR. DOLAN: Yes, I'm here. Can you hear me?

MR. PEEBLES: Yes. Can you provide a correction to an earlier statement about lithium impurities?

MR. DOLAN: Yes. So a couple of slides ago, we were talking about which sources of tritium are included for these calculations on tritium MAR for the MHA. So in our initial analysis here, we are just including the tritium sources produced by neutron irradiation of Flibe, so the numbers fed to the MHA in the current state do not include evaluations of tritium produced by lithium impurities in the graphite. We don't expect those to be significant contributors to the overall tritium production or tritium source term, but that is a detail we could evaluate for source term tritium calculations in the operating license application.

CHAIR PETTI: I think you're right. It's

just worth confirming it when you get to the OL.

MR. DOLAN: Right.

MR. DENMAN: Thank you very much, Kieran. My apologies on that misstatement. Okay. So for argon-41 released, argon-41 is primarily produced via neutron activation of argon-40 to argon-41, and we are assuming that the inventory available for release from our system consists of the argon-41 contained within the graphite's closed porosity.

Okay. Next slide. For our release models, we will talk first about our TRISO fuel. The time-temperature history for this fuel, and this fuel is in the in-core fuel or, you know, submerged within the Flibe, you can see the time-temperature history as pointed out. It's, first, this higher dotted line, and then it moves into the more solid darker line. All of the fuel within the core is assumed to be at this temperature simultaneously.

Transport through the TRISO layers are modeled using fixed law of diffusion. The CORSOR model is used for kernel diffusivity or diffusion of radionuclides out of the kernel. And then the IAEA correlations described in the construction permit application are used for layered diffusivity or movement of radionuclides through each of those layers.

Diffusion, again, is driven by this hypothetical temperature curve. Each layer and the kernel all have the exact same temperature as described this temperature curve. And transient diffusion of fission products was shown, given the low temperatures that we see in this hypothetical temperature curve, even though they're bounding of our postulated events, they're low for TRISO accident analysis in general, is negligible if even a single pick layer remains intact. Thus, the total release from our fuel is really dominated by releases from exposed kernels within the TRISO configuration.

Okay. Next slide. The --

MEMBER KIRCHNER: This is Walt Kirchner. Could I ask, could you -- I'm not sure this is a proprietary because you're using EPRI's spec. What's your assumption on the exposed kernel fraction? Because you're right. At these temperatures, that would be the dominate source of uranium and/or fission products.

MR. DENMAN: I'm not sure if that number is proprietary. Let me look to my --

CHAIR PETTI: I hope not. It's in our letter. It's been in our memos.

MEMBER KIRCHNER: Yes. It's in the EPRI spec, if that's what you're using.

MR. DENMAN: It's very close to that spec.

MEMBER KIRCHNER: Can you hazard a number?

CHAIR PETTI: Yes, aren't you assuming a much higher number than EPRI?

MR. PEEBLES: So we can confirm that it's not proprietary and then get back to you after the break, if that works.

MR. DENMAN: I'd want to look up the exact number. I don't have it right in front of me, but we can get back to you.

Okay. So maximum hypothetical releases from our Flibe coolant. The Flibe provides a secondary functional containment barrier bounding, this bounds the circulating activity or, sorry, Flibe provides secondary functional containment barrier to both the bounding circulating activity and our in-transient releases of fission products from TRISO.

There are two primary release pathways from the Flibe. These include bubble burst as the initial assumed conservative void fraction, bursts at the top of our Flibe free surface, and then the evaporation, which is driven by the time-temperature curve.

Certain radionuclide groups effectively

bypass Flibe's functional containment as no credit is given for gas retention within our Flibe in the MHA, and highly-volatile noble metals who have a high-vapor pressure or are modeled as having a high-vapor pressure evaporate extremely quickly in our MHA. Thus, they effectively have no hold up.

CHAIR PETTI: So, Matt, just to be clear since most of those fission products aren't that important, iodine is like a noble gas that follows that pathway?

MR. DENMAN: Per our mechanistic source term topical report, iodine is grouped as a salt-soluble fluoride.

CHAIR PETTI: Oh, okay. So it's like cesium. It stays in the salt.

MR. DENMAN: Correct.

CHAIR PETTI: And then has -- okay.

MR. SCHULTZ: Matt, Steve Schultz. You sort of mentioned this before, but, the bounding circulating activity, you assume what is in technical specifications for that value in the calculation?

MR. DENMAN: Yes. Thank you very much for the question. The circulating activity is assumed to be maintained via technical specifications, although those values will not be provided until OL.

MR. SCHULTZ: But in the numbers that you provided in the MHA calculation, you depict a typical number that might be used in the technical specifications or just bounded it in some fashion?

MR. DENMAN: We bounded what we believe to be, what would be in the circulating activity given the state of the design as reflected in the PSAR.

MR. SCHULTZ: Okay. Thank you.

MEMBER KIRCHNER: This is Walt Kirchner. You would then, Matthew, do the same thing with the argon cover gas, right? Because on the previous slide, you talked about argon-41 release that had been trapped in structure, but the cover gas would be activated, as well. So that would be controlled by tech specs, and that would be added into the MHA?

MR. DENMAN: Yes.

MEMBER KIRCHNER: Thank you.

MR. DENMAN: Okay. So I think we can move on to the next slide. For structural MAR, tritium is assumed to be held within the graphite grains. No hold up of tritium, and the Flibe instantly drops the concentration of tritium outside the graphite grains to zero. So, effectively, the grains are modeled as a sphere. You have a constant flux of tritium that's pushing more and more tritium into that graphite grain. The flux drops to zero outside of the grain, and now all of that tritium that was being forced into the grain is now able to rapidly diffuse out of the grain due to this immediate and non-physical concentration gradient.

The MAR outside of the graphite grains are instantly released at the start of the transient; and within tens of hours, basically, all of the tritium that is stored within these grains are modeled to be released, which is non-physical and extremely conservative.

Next slide. So then we can move on to our gas and atmospheric transport. Once you have any gases and evaporated materials that leave our functional containment, they bypass the vessel head and go directly into the reactor building. That's what they're modeled to do. In reality, the vessel head would contain these radionuclides, but they're modeled to bypass the vessel head. And then they're input into RADTRAD. RADTRAD has two depletion mechanisms that we use for radionuclides that enter the reactor building. That is radioactive decay and aerosol settling through the Henry correlation. There is a conservative two-hour hold-up assumption applied to radionuclides that enter the reactor building, and, after that two-hour hold-up assumption is applied, they are released to the environment. This is where ARCON 96 is used to calculate our dispersion estimates, our chi over Qs. It inputs hourly radiological data. It evaluates distances from the reactor building to the exclusionary boundary in low-population zone, and it uses multiple approved values from the KP-TR-12 topical report. And once all that information is fed in, out is provided the time average dispersion values which you can see on the table.

Next slide.

MR. SCHULTZ: Matt, Matt, this is Steve. I'm sorry. Are you finished here?

MR. DENMAN: Yes.

MR. SCHULTZ: If you didn't assume any depletion mechanisms, how much would that affect your answer for release in RADTRAD?

MR. DENMAN: Steve, thank you for your question. I believe, as part of our methodology, we always look at the release, we always calculate the release fraction from the building. Those release fractions are, the equation for the building release fraction that we propose is in our mechanistic source term topical report. I believe, and it's been a while since I've looked at it, but it's roughly on the order of a release fraction of 0.9-ish.

It's a little different for different radionuclides. They have different decay rates. Gases, obviously, don't settle. Our off-site releases are heavily dominated by gases, but it's roughly on the order of 0.9.

MR. SCHULTZ: That makes sense. Thank you.

MEMBER KIRCHNER: May I ask a question? This is Walt again. Matthew, since you don't take credit for confinement, when you look at the leakage from the reactor building, I presume it would be at the upper level, not the ground level. Did you look at how that might impact your results?

MR. DENMAN: Thank you very much for your question. Can I restate it, restate your question to make sure I understand?

MEMBER KIRCHNER: Sure.

MR. DENMAN: You're asking, I believe you're asking did we look at the delta between an elevated release and a ground-level release to see what the dispersion changes would --

MEMBER KIRCHNER: Yes, that's one part.

MR. DENMAN: -- or how that would impact dispersion changes. Thank you for that question. No, we did not model the difference between an elevated release versus a ground-level release due to the approval of the ground-level release being conservative in our mechanistic source term topical report.

MEMBER KIRCHNER: Okay. I can go back and review again the mechanistic source term topical report. But then a second question is do you have separation. A two-hour hold up, that's based on, if I remember correctly, that's based on civil engineering code standards for unventilated building, but you're dealing with hot, potentially hot gases or at least a fairly warm environment, and you're dealing with tritium. Does that factor into these analyses?

MR. DENMAN: Thank you very much for that question. The two-hour hold up, again, is a parameter that was approved within a mechanistic source term topical report. It was actually pulled from NRC guidance for design basis accident dose calculations from fuel handling accidents in the spent fuel building and releases in open containment. So if you're doing, if you're moving fuel within the containment of a light water reactor, you have the doors open, and you have a release of radioactive material and it's just allowed to migrate out of an open building, that's where the two-hour hold-up assumption came from, and that was the basis for our argument within the approved source term topical report.

MEMBER KIRCHNER: Okay. Thank you.

MEMBER HALNON: Just one other question. This is Greg. The ARCON 96, did you do any sensitivity runs on that based on different site layouts? In other words, different buildings may be in the way versus a clear path to the site boundary?

MR. DENMAN: Thank you very much for that question. No, we did not do any calculations of a torturous path of the plume through the building. We used the straightest path from the exterior of our building to the site boundary.

MEMBER HALNON: Do you feel like that's the most conservative, given the potential wave effect of different buildings that may be in the way that could actually cause a redirection of different air flows?

MR. DENMAN: Yes, we believe that that is the conservative path.

MEMBER HALNON: Okay. Thanks.

MR. DENMAN: Okay. So now to the results. As is seen on this table, the dose results meet the 10 CFR 100.11 reference values at the EAB and LPZ with significant margin both for the exclusionary boundary, whole-body dose, and thyroid, and the low-population zone over 30 days.

Okay. Next slide. To conclude, the MHA dose consequence results meet the site dose reference values in 10 CFR 100.11(a)(1) and (2) at the EAB and LPZ with significant margin, and the MHA dose is bounding because it employs various non-physical conditions that are beyond the expectations of design basis calculations.

And with that, thank you very much for your time, attention, and questions.

CHAIR PETTI: Members, any additional questions?

MEMBER KIRCHNER: How are you going to proceed, Dave? Are we going to hear from the staff on MHA, or are we going to events next?

CHAIR PETTI: MHA first, I think.

MEMBER KIRCHNER: Okay.

CHAIR PETTI: And then tomorrow will be the accident detail. So then why don't we hear from the staff. Is it Michelle? Yes.

MS. HART: Good afternoon. I'm Michelle Hart from the staff; I'll just say that.

(Laughter.)

MS. HART: I forget who I work for. We're here today to talk about the staff's review of the preliminary analysis of the maximum hypothetical accident for the Hermes PSAR.

Next slide, please. Okay. So Kairos just provided a thorough description of the maximum hypothetical accident assumptions, methods, and consequence analysis, as described in the PSAR. As they described, the MHA describes a hypothetical radionuclide release intended to result in consequences that are bounding for the postulated events.

With respect to the MHA as bounding, PSAR Section 13.2.2 described the postulated event methodology and the figures of merit and acceptance criteria that Kairos developed to provide assurance that the MHA consequence analysis is bounding for postulated events, and we'll be describing our evaluation of that information at tomorrow's meeting.

There are a couple of referenced topical reports that are relevant to the MHA analysis, and that is the fuel qualification methodology and the mechanistic source term methodology.

Next slide, please. We had a lot of discussion about the MHA hypothetical temperature versus time profile that is given to give bounding radionuclide releases for the MHA. As Kairos had described, it's not a specific scenario. It's not physical. And because fission product release and transport is mainly through diffusion driven by temperature, it would maximize the releases. And final determination of that temperature versus time curve is conservative for the postulated events will be done during the operating license review.

As Kairos has described, it assumes that the safety related systems function as designed but includes consideration of the single failure criterion, even though it's not directly modeled in the MHA analysis and there are no incremental fuel particle coding failures from the transient.

Next slide, please. So for the consequence analysis, they do refer to the accident source term methodology that was in the approved topical report. It models the system as sources of radioactive material at risk of release or MAR and the barriers to release. They apply a release fraction to each barrier to eventually result in release to the environment, and that's consistent with the description of a functional containment. And they do also model gravitational settling of Flibe aerosols in the reactor building consistent with the approval in the topical report.

Next slide, please. To go to a little bit of our evaluation of the MHA source term modeling, as we've described several times, that temperature-time profile does drive the diffusion releases from fuel, Flibe, and graphite. The MHA assumes conservative fuel, Flibe, structural and cover gas releases. In effect, the complete fuel inventory is available for release into the Flibe. The bounding failed fuel fractions by cohort are assumed. That's the different particle layer of failures and bare particles, as well. Flibe and cover gas radionuclide inventories are set to technical specification values which will be provided at the OL.

Except for the fuel transient releases, tritium and argon-41 modeling, the MHA uses approved mechanistic source term models from the topical report. The fuel releases are modeled using accepted methods, and the staff reviewed the fuel release references to find those models acceptable.

The tritium modeling that they have in the MHA resulted in higher total releases than would be expected from the topical report methodology, and the staff also evaluated the modeling assumptions for both the tritium releases and argon-41 in the audit, and we found them to be conservative.

Next slide, please. As noted before, the mechanistic source term topical report methodology will not be fully implemented until it's used by the applicant in the operating license application FSAR. And the staff will review the final implementation of that topical report for the Hermes, including the limitations and conditions in the topical report SE in its review of the operating license application.

Staff presents its evaluation of the site characteristic accident atmospheric dispersion factors to the subcommittee on March 23rd.

Next slide, please. So to go into some of the audit, some of the information that we audited. We did look at the preliminary consequence analysis and MHA source term information. So we did see their calculation packages, output from codes, things like that.

In the audit, we were able to confirm the PSAR description of their MHA analysis. In those calculation and reference reports supporting those calculations, we were able to see how they determined the initial radionuclide inventory and MAR sources, including for fuel and Flibe, and those calculations of tritium generation and argon-41 inventories, how they modeled those in the graphite. We were able to see the calculations estimating releases from the graphite and the modeling of radionuclide transport across barriers and the release fractions for those barriers, as well.

We also were able to have an in-person discussion with the staff that they could show us how they went through that process using cesium as an example, isotope, to show us how they could actually put it into the RADTRAD code to generate the doses.

MR. BLEY: Michelle, Dennis Bley with a question. It actually goes back a slide.

MS. HART: Okay.

MR. BLEY: But conservative is a word that makes me a little nervous whenever I hear it. Can you talk a little bit about what you found conservative? Was it the results in the quantity released? Was it the models? Was it the assumptions? Where did you find the conservatism?

MS. HART: So the majority of the conservatism that we really had and in the discussions with the staff at Kairos was there was a lot of conservative-leaning assumptions. They made bounding assumptions. You know, we were able to see that they had used appropriate models or models that we were aware are appropriate for the use.

MR. BLEY: So if I understand you right, given the assumptions, you think the modeling was reasonable, but it's the assumptions that you found to be conservative?

MS. HART: Do you want to add something to that, Jeff?

MR. SCHMIDT: Yes. This is Jeff Schmidt from the staff. So, you know, Matt kind of laid out a bunch of conservatisms as he went through there. So it's things like graphite being a perfect absorber and then the release fractions from that graphite. Like, they looked at different diffusivities to maximize that release. Pebble release fractions were -- I hope this isn't proprietary -- were near one at those temperatures. So the mass transport of tritium into the graphite was a conservative calculation. The fact that the fuel inventory, nothing was allowed to leak away while the coolant activity is also at its tech spec value is a conservatism.

So I think it's hard to break out, like, single -- there are multiple levels of conservatism in this calculation. And I think a lot of those were just covered.

MR. BLEY: Okay. Thanks.

CHAIR PETTI: Michelle, do you remember what isotopes dominated the dose?

MS. HART: So from what I remember, it was mostly tritium and argon.

CHAIR PETTI: Okay. That's what I would have expected. That's what my gut said.

MEMBER MARCH-LEUBA: If you went on the blaming game and you have to blame somebody for how low these numbers are, would you blame the fact that, and, by blame, I mean the fact that the various fractions of TRISO fuel has failed, and the fraction that has not failed does not raise anything. Is that why we're getting these ridiculously low numbers with these conservative assumptions?

MS. HART: So I would say it's fair to state that the TRISO particles are retaining the majority of the fission products. Flibe does retain some. Did we look at specific failure fractions and did they provide sensitivity analysis on that? No, not at this stage.

MEMBER MARCH-LEUBA: What gives me comfort when I look at this design, it's not that they assume various fraction of particles that are failed but that they measure it when they operate by measuring the contamination in the activity of the Flibe.

MS. HART: Right.

MEMBER MARCH-LEUBA: So if we start operating and suddenly we see a hundred times the Flibe, we'll stop and we'll figure out what's going on. So the fact that it's something that we're measuring and we can know what it is is good.

MEMBER KIRCHNER: Jose, this is Walt. If I could observe, at the time-temperature curve that they're using for the TRISO fuel, the fuel meets the spec. You're hardly challenging it. So as I think Michelle answered, it's going to be tritium and argon because you're not assuming the actual produced fuel performs that well. That's the reason why the numbers are so very, very low.

Now, as you said, if they have a batch of fuel that turns out not to be up to spec, they'll see it right away in the circulating inventory and in the cover gas. You'll see that almost instantly if there's a large, a much larger defect fraction for kernels and particles that are either defective or there's tramp uranium outside of the particles.

CHAIR PETTI: Those are not the most difficult QC techniques. If you get bad fuel, you know it in QC. It's pretty obvious. You know, you get zero, and then you get integral, most of the time, integral measurements of kernels. So, oh, that pebble has three exposed kernels, that one has none. It's very clear when you do the test.

MEMBER KIRCHNER: On that subject, I guess, you're more familiar with this. It's been a long time since I've looked at these equations for the TRISO particle performance, but I would submit, at these temperatures, you're not going to see much of an impact, assuming, again, the fuel meets the spec.

CHAIR PETTI: These temperatures are so much lower than you have in an HTGR that the diffusion --

MEMBER KIRCHNER: From a calculational standpoint, you're not going to see anything using the approved equations, methods, for analyzing TRISO performance.

MEMBER MARCH-LEUBA: The dose at the perimeter of the plant is controlled by your fabrication. You don't make any mistakes, and that's easy to quality control. It's reliable.

CHAIR PETTI: And the fact that they assume in-service failure, normal operation failure, a hundred times with the AGR program demonstration.

MEMBER MARCH-LEUBA: It's a good margin,

but we still haven't operated a reactor. Let's get a couple of years of running it and see what happens.

MS. HART: All right. So the only other thing I wanted to say about this is, because we did have this extensive audit, there was no need for me to do a consequence analysis, a confirmatory analysis.

Next slide, please. So our evaluation findings were that we do find that the MHA serves as a bounding hypothetical analysis for the Hermes reactor. The combination of bounding conditions analyzed are beyond what is assumed for postulated events. The preliminary dose analysis for the MHA are subsequently below the regulatory dose reference values for test reactor siting in 10 CFR 100.11. And because the assumptions of the MHA are bounding, calculated doses would likely not be exceeded by any accident considered credible and the staff will confirm calculations as part of the OL application review.

Next slide, please. We did have to talk a little bit about control room habitability. It was really described in PSAR Section 7.4. They did not provide a dose analysis or design details for control room radiological habitability in the PSAR. However, we expect that they will do some kind of analysis to show compliance with PDC-19. They did identify that as the relevant design basis for control room habitability design, and an additional description of the control room habitability design and dose analysis corresponding to the final design will be provided in the OL application.

MEMBER MARCH-LEUBA: This is Jose. Maybe because we have such a serious concern about the assumptions in our numbers, but the release inside the plant is very large. Everything goes in there. So habitability with this conservative analysis may be an issue that you exceed applicable doses. Doses for tritium are really, really low, and that will apply mostly to the reactor areas but they move to the control room, too.

MS. HART: Yes, it is certainly something that we have in our sights to evaluate in the OL application when we do that in the shielding analysis and any further --

MEMBER MARCH-LEUBA: Yes, the steady state you can release in the normal operating because the temperatures are so high that it's going to leak like a sieve.

CHAIR PETTI: But the assumptions that they've used are very cavalier, shall we say, because it still meets the off-site dose. But when you start talking about worker safety, they're going to need a sharper pencil, and I'm sure they will. It will be clean-up systems. It's probably going to be a very different sort of look than what you see in the PSAR.

MEMBER KIRCHNER: And we talked a little bit about that at the March meeting, as well.

CHAIR PETTI: Yes.

MS. HART: Next slide, please. And so, in conclusion, the NRC staff does find the preliminary design information and analysis are consistent with the applicable criteria in NUREG 1537 and that we conclude that the information on the MHA is sufficient for the issuance of a CP, and any further information can be reasonably left for OL application.

Are there any further questions?

MEMBER KIRCHNER: Michelle, this is Walt Kirchner. I know we've got the groups of events that were analyzed as part of Chapter 13 coming next or coming tomorrow. When you, the staff, went through the applicant's selection of events that they thought were limiting, did you flag any in particular that you would be concerned about and want to go back and re-examine whether or not they might, any of those individual events might challenge this MHA assumption?

Because, basically, as we've been discussing, this MHA doesn't really involve any significant release from the fuel.

MS. HART: Nor does it really look at oxidation of exposed graphite. We kind of mostly were thinking about doing comparisons to the salt spill and the pebble handling system failure. I don't know, Jeff, if you had some additional thought on that.

MR. SCHMIDT: Yes, this is Jeff Schmidt. I just want to echo what Michelle said. So I did a lot of the evaluations for what I would call the dose accidents in Chapter 13 for the postulated events, and she's right. Those are the ones I kind of were constantly questioning whether the MHA would bound those because I really didn't have a great engineering feel for how much salt is spilled, what's the release from the salt, what's the aerosol generation from the salt spill. So, you know, I used some of their illustrative examples in the appendices of KP-TR-018 to get some sense for it. I asked for some temperature profiles, what was holding the heat in a salt spill, how much would I heat up due to a salt spill accident, for example, to threaten those temperatures of the MHA.

So those are the accidents, the salt spill and the PHHS event, were the ones I kind of focused on.

MEMBER KIRCHNER: Okay. Those are the same two that I have concerns about, the potential for any of the pebbles to be exposed in the pebble handling machine or system. Until we see the detailed design, it's hard to know where the level will wind up in the reactor vessel. Certainly, there's discussion from the applicant of unmitigated air ingress. How much graphite is exposed is going to be a design detail, I suspect. Could it result in any of the pebbles being uncovered by Flibe would be something of concern, as well.

Okay. Thank you.

CHAIR PETTI: Any other questions, members? Okay. Well, we're well ahead of schedule. I just think we should keep pushing through. Are you ready to talk about the other part of Chapter 13 today?

PARTICIPANT: Yes, we are.

CHAIR PETTI: Okay. We can do the break early. I had it circled at 3:10. It's 2:40. Okay. Then let's take a break until 3:00, and then we'll come back and we'll start the other sections. Thank you.

(Whereupon, the above-entitled matter went off the record at 2:41 p.m. and then went back on the record at 3:00 p.m.)

CHAIR PETTI: Okay. We're back and ready to start with Kairos. Matthew.

MR. PEEBLES: This is Drew Peebles, Senior Licensing Manager. Just before we get started, we were talking about the exposed kernel fraction. We did check, and that is marked as proprietary in the topical report. But in our fuel qualification methodology topical, KP-TR-011, it's Table 313, if that helps. But I can say in the public session that the fraction that we assumed is not less conservative than the AGR 2 spec.

CHAIR PETTI: Okay. Yes, we had a side discussion and came to the same conclusions. Thanks.

MR. PEEBLES: Okay. I'll turn it over to Matt. Thank you.

MR. DENMAN: Okay. Well, thank you. So my name is Matthew Denman once again, and thank you very much for the opportunity to talk to you about Chapter 13 accident analysis focusing on postulated events.

You will see these next two slides are a little bit of repeat from what you heard earlier today. We were expecting to give these tomorrow morning, and we wanted to provide context again. But just as a refresher, 10 CFR 50.34(a)(4) does require a preliminary safety analysis to assess the risk of public health and safety from operations of a facility, including determination of margins to safety. I won't go too far into the MHA again, other than that this is, the MHA is supposed to bound postulated events and it is analyzed for dose compliance with 10 CFR 100.11.

The list of postulated events are comprehensive to ensure that any event with a potential for significant radiological consequences has been considered. Initiating events and scenarios are grouped so that the limiting case for each group can be qualitatively described in the CPA, and acceptance criteria are provided for important figures of merit in each postulated event group to ensure that potential consequences of that event group are bound by the MHA as the design progresses. Additionally, prevention of initiators are justified in the PSAR.

If we go to the next slide, again, this is a conceptual slide to show the relationship between the 100.11(a)(1) and (2) reference values. The MHA and the potential postulated event doses where the MHA demonstrates your margin to the reference value and then the hypothetical natures and assumptions and boundary conditions and models within the MHA provide that stand-off between the MHA doses and the potential postulated event doses.

Next slide. So getting into the postulated event analysis methodology, postulated events are identified in Chapter 13 of the PSAR. Postulated events include any potential upset of plant operations within the design basis that causes an unplanned transient to occur. Justification is provided for those events excluded from the design basis. Figures of merit are provided or, sorry, figures of merit provide the means to measure and demonstrate the resulting doses from postulated events are bound by the doses of the MHA.

The preliminary methods and sample calculations of postulated event groups are provided in KP-TR-18, Rev 2. This methodology describes how analyzed figures of merit for, how the figures of merit for each postulated group are analyzed and how acceptance criteria will ensure proper mapping between the off-site dose consequences of the postulated events and the MHA which bounds those events.

The final safety analysis results will be provided with the operating license, including verification and validation of the evaluation models that will be used.

So for the next slide, I'm going to transition to my colleague, Tim. Tim, please introduce yourself.

MR. DRZEWIECKI: Thank you. This is Tim Drzewiecki. I'm a safety analysis manager here at Kairos Power. I'm going to spend a few minutes talking about a postulated event analysis methodology.

So we do follow the steps that are outlined on the in-depth process Reg Guide 1.203. Some of those elements are discussed in our technical report KP-TR-18. Postulated events with similar characteristics are grouped into categories which is consistent with NUREG 1537. Limiting event in each category is then identified and, again, qualitatively assessed from the event initiation until a safe state is reached. That safe state is defined in the methods for each event category as the point where the transient figures of merit have been stabilized in a safe condition and generally involves things like, you know, some criticality and decay heat removal.

Next slide, please. As far as the inputs for the postulated events analyses, these are actually shown in Table 44 of KP-TR-18. There are 15 parameters in total. Some of them are biased in a conservative direction. Some are nominal. But several are also varied over a range, and the bases for these are described in that table.

Again, a range of values are assessed to identify a limiting scenario for each postulated event and key modeling uncertainties and initial conditions are applied to the methods to ensure that the figures of merit are conservatively predicted. And those figures of merit again are shown in Table 13.11 of the PSAR.

Next slide, please. So I was going to hit a couple of events, and then I'm going to just kind of walk through what a typical event is going to look like in our reactor. So for the loss of forced circulation, the limiting event here was a pump seizure that would disable primary salt pump, and in that event is we do see is a heat up of the system which is then detected by the protection system. That causes a trip early in this event.

And then other events that are predicted here are things like a pump trip or a loss of normal heat sink. The next category is the insertion of excess reactivity. This is a control system or operator error that causes an element to withdraw continuously at the maximum speed, and this is detected by the protection system either by a high flux or a high temperature. Events that also fall under this category are errors in fuel loading, reflector shifting, or venting of gas level. And, last, a category I'm going to cover on this slide is general challenges to normal operation. So this would be any kind of challenge to operation that's not covered by the other event categories. We think these are bounded by the loss of poor circulation, and they include things like spurious trips, operator errors, and equipment failures.

So this next slide, I'm going to walk through just a loss of forced circulation overheating event. Now, those images that you see are actually the same image or at least the image on the right. That's adapted from a figure from KP-TR-18 just so who the time scale a little more clearly because the one on the right goes out to about 72 hours and is on a standard scale, as opposed to a semi-log scale.

So this event starts with a pump seizure or a locked rotor. We do see the heat-up that occurs in the first minute of this event at about 30 seconds, and that would show up, on the left is one of those peak lines there. We do see a reactor trip. And then following that, there is a heat-up period in which our decay heat is higher than the heat that's being pulled out by our DHRS, or decay heat removal system. That heat-up period lasts for about 20 hours, at which point the heat removal from decay heat removal system exceeds our decay heat loads and then we see a decrease in the system temperature.

So if there's no questions, I'll hand it back to my colleague, Matt.

MR. DENMAN: Well, thank you very much, Tim. So I'm going to cover some of the postulated events that really involve releases of radioactive material outside of the vessel.

So the first event is the mishandling or malfunction of the pebble handling and storage system. The limiting event involves a break in the fuel transfer line during removal of fuel from the core that results in a spill of pebbles within the transfer line into the surrounding room. The reactor protection system detects this condition and initiates a trip of the pebble handling and storage system to prevent additional pebbles from moving into the pebble transfer line. Grouped events include transfer line breaks when pebbles are inserted into an empty core, core at power, storage canisters, and mishandling fuel outside of the reactor.

MEMBER KIRCHNER: Matthew, this is Walt Kirchner. Just quickly, you say the reactor protection system detects this condition. What would be the sensor for that? Gamma detection?

MR. DENMAN: Pressure.

MEMBER KIRCHNER: Well, you wouldn't detect it on neutrons from your core flux monitoring system. So is the idea that in the reactor cavity you would have a sensor?

MR. DENMAN: So this would be a pressure-related trip on the cover gas system.

MEMBER KIRCHNER: Well, it's pretty low pressure. Okay. Okay. Thank you.

MR. DENMAN: Okay. I will also note here that the pebbles themselves do have a low decay heat level and, thus, temperatures will be manageable.

The radioactive release material from a subsystem or component, the limiting event is assumed to be a seismic event that results in the failure of all systems containing radioactive material that are not qualified to maintain structural integrity during a design basis earthquake. This is effectively a common mode failure. Design requirements on the amount of MAR for these structure systems and components will be set to ensure that the amount of

MAR that could be released is less than the MAR derived from the maximum hypothetical accident releases. And grouped events include releases from the tritium management system, inert gas system, chemistry control system, and inventory management systems.

Next slide. Salt spills. So in this scenario, a hypothetical double-ended guillotine break occurs in the primary heat transport system hot leg piping. The reactor protection system detects the salt spill due to a low coolant level and initiates a reactor trip. The grouped events for this scenario include spurious draining of the primary heat transport system, leaks from other Flibe-containing systems, mechanical impact or collision of Flibe-bearing structure systems and components, and heat rejection radiator tube breaks.

Finally, internal and external hazards are considered. These include internal fires, internal water flood, seismic events, high wind, toxic releases, mechanical impacts or collisions, structure systems and components, and external floods as described in Chapter 2 of the PSAR. Events in this category are bound or considered as initiators to other event categories. A good example of this is the release of radioactive material from subsystems or components that I talked about on the previous slide where an external hazard provides that common mode for release pathway for all of those.

So in conclusion, postulated events within the design basis are identified and grouped by characteristics and modeling approaches used to evaluate these postulated events. Design features which are credited with mitigating the effects of postulated events are described. Figures of merit are derived for the postulated events to provide surrogate metrics which demonstrate that the resulting doses are bound by the dose consequences of the maximum hypothetical accident analysis. The acceptance criteria for these figures of merit represent design limits that ensure that the MHA will remain bounding.

And with that, I appreciate the ACRS for their attention and questions. And thank you.

CHAIR PETTI: Matt, I had a question. It wasn't clear to me in some of the events whether the single failure criteria is applied or even has to be applied in these events, particularly in core sort of events.

MR. DRZEWIECKI: Yes, Dave, this is Tim Drzewiecki. And, yes, we do apply single failure criteria. We do, you know, account for a stuck rod, as well. The single failure is generally associated with our decay heat removal system. That's generally seen to be our limiting single failure.

CHAIR PETTI: But let's look at the reactivity event. Is there a delayed detection? You say that the high flux is out, but then the higher power gets you, shuts it down? What's the timing there?

MR. DRZEWIECKI: The timing. So in terms, those specific, you know, like, details, in terms of what trip would come in then, those would have to be, you know, looked at. But the one thing I do want to highlight is in terms of our, you know, RPS is designed to, you know, be single failure-proof or to actually handle single failures. You know, that's accordance with the standard that it's designed to.

CHAIR PETTI: So you think that the event that's modeled in the appendix of the technical report is still fairly reasonable once you get the final design details? You're not going to see a greater response, if you will.

MR. DRZEWIECKI: I can't speak to that because our methods are still being developed. You know, those calculations were based on preliminary design information, so I think it's representative of what we're going to see. But I can't say that it's bounding. There are things in there that are very conservative. For example, reactivity insertion. Those are very conservative, so it could be bounding but I can't commit to that.

MEMBER HALNON: This is Greg. Pardon me if we've talked about this. The occupational dose with the RBHVAC, I assume that you're assuming that, since it's non-safety, it's essentially not there. Is that another analysis another time, or is it factored into this MHA?

MR. DENMAN: So occupational dose evaluation will be provided at the OL.

MEMBER HALNON: And just surmising that this MHA is going to exceed any occupational dose allowables, what happens then? Do you have to come back and re-look at the MHA, or do you have to design something into the RBHVAC to control the environment better?

MR. DENMAN: The MHA is intended to analyze off-site doses, not occupational doses.

MEMBER HALNON: Okay. So that will be, this will be unaffected by any inside dose, if you will.

MR. DENMAN: Correct.

MEMBER HALNON: Okay.

CHAIR PETTI: Any other questions, members?

MEMBER REMPE: To follow up, on page 47 out of 99, you talk about the spilled pebbles, and you say since the temperatures are high they'll react with the air in the building to generate heat because it's an exothermic reaction, and I just wondered do you know what temperature they're at?

MR. DENMAN: Thank you very much for that question. So the pebbles are, by the time they actually make it out of the core and make it into the cover gas space above the Flibe free surface, they're going to be very, very, very close to the cover gas temperature because the decay heat is so low and the pebbles are fairly small. As they move through the pebble handling and storage system, that trend is going not follow. So as you get the temperatures in the pebble transfer line, the temperatures of the pebbles are going to start to decrease. And then in a spill event, they're assumed to still be above the 400 C oxidation threshold temperature, but it's not expected to be a rapid process, nor at a process where you're likely going to see exothermic temperatures. It will likely be endothermic. But, again, these are all preliminary design information feeding into these temperatures, and we'll have to look to OL to know for sure and we will be ready to evaluate any condition that we find.

MEMBER REMPE: So I think the answer is that you're not exactly sure because you're modeling hasn't progressed that far, right, is what the answer is? Because I didn't hear a temperature really coming out.

MR. DENMAN: Yes. So --

MEMBER REMPE: Okay. Thank you.

MR. DENMAN: Okay.

CHAIR PETTI: Okay. Hearing no more comments, let's move to the staff. Jeff?

MR. SCHMIDT: Jeff Schmidt with staff. I'll wait for my slides.

Okay. So we're going to talk about the same things that Kairos just got done talking about, postulated events in other sections.

Next slide, please.

Kairos, as we talked about, uses the MHA. The MHA is supposed to bound the radiological release, and there has been some reference to this PSAR Table 13.1-1, which I think is worth bringing up again, because what that table is trying to communicate is that different events have different release pathways, and you've got to control some of the variables or the figures of merit in the table to ensure that the MHA remains bounding.

I think we've said that multiple times, but it's important to understand what the -- what the purpose of that table is.

Postulated events considered are consistent with those listed in 1537, as Tim just said. Though there were some technology-specific events or event sequences that are precluded by design, we'll talk about two that the staff had additional questions on.

And, obviously, we've talked about these in previous meetings. The Flibe interaction with water or concrete are precluded by design, and that's listed in that PSAR section.

Some technology-specific events such as increased pebble packing fraction and the potential reactivity insertion due to that have been evaluated, at least to the design information available.

Next slide, please.

As we talked about, the postulated event methodologies in KPTR-018 Rev 2. As Tim mentioned also, KPTR-018 Table 4 has input parameters, which kind of outline the overall methodology that's going to be applied to the postulated events. It covers things like initial power level, reactor coolant temperatures.

We spent some time with Kairos flushing out the details of that to ensure it was a very I think thorough and consistent overall calculational framework, relatively along the lines of, let's say, like a NUREG-0800 Chapter 15 analysis.

FSAR analyses will consider the full range of sensitivities based on the Table 4-4. KP-SAM and KP-BISON have the capability to model postulated events, corresponding fuel releases. We talked a little bit about that in our previous meeting and the capability of those codes. Just to remind everybody, code verification and validation will be reviewed prior to or as part of the OL application.

Next slide, please.

So I'm going to walk through each one of the events kind of the way they're listed, the way -- I'm sorry, the way they're listed in the -- in the PSAR. So the first one is insertion of excess reactivity. Seems to continuously draw the highest worth control rod at the maximum speed. Reactor trips on high power or high temperature. Range of reactivity insertion rates and initial core power levels will be evaluated at the OL.

So right now they've done the max -- like a maximum reactivity insertion, but usually you look at a different range of insertions because different trips will pick up different reactivity insertion rates.

Uncertainties will be quantified as part of the OL application. Internal element injection is precluded due to the low differential pressure between the reactor and atmosphere, so that's a consideration in, you know, what the events are for -- that are considered as part of insertion of excess reactivity.

Temperatures stay below the MHA, hypothetical temperature versus time curve, except for the maximum reflector temperature, which slightly exceeds the MHA-free surface and graphite temperature limits for a short period of time.

Again, you know, it's important to stress that these are preliminary calculations. At short deviation was considered by the staff in that review and thought to -- that the MHA was still going to be bounding because it's a fairly short duration and a relatively small deviation from the acceptance line.

Staff scoping analysis yielded similar results, as we show in the following slides. So we're going to have -- at the end of this presentation, we're going to go through some of the scoping analyses that were performed by the staff, and basically go through a comparison of their calculations to our calculations. Andy Bielen will be handling that.

The staff has reasonable assurance that the MHA dose bounds that of the insertion of excess reactivity because of conservatisms in the MHA analysis. As we talked about, there's a number of conservatisms in the MHA analysis. There is no real separate or different pathway to exposure here, say like for the pebble handling system or the salt system. So that was how we reached the conclusion that the MHA was going to be bounding, just based on the temperature profile that's used as part of the MHA.

Next slide, please?

So the salt spill is the next postulated event. This is a loss of coolant inventory resulting in different release pathways in the MHA. As was stated earlier, some safety-related systems work as intended, assumes water or concrete interactions are precluded by design. That's really referring to, you know, the -- where the salt is spilled.

Methodology includes evaluating a range of break sizes and locations as part of the calculational framework. As Matt described, their day-to-day, double-ended guillotine break of the hot leg.

Release pathways, different from the MHA, include radionuclide by the break, evaporation from the spilled fuel pool, and oxidation of any exposed graphite. And we've talked a little bit about that as -- you know, right now we have preliminary estimates of like the amount of salt spilled, but how much graphite that is exposed during that transient the staff is not sure of yet.

But that's one of the figures of merit that has to be controlled, is that, you know, you have to limit the oxidation such that, you know, oxidation doesn't release or doesn't lead to, you know, contributing to a release that's greater than the MHA.

Heat-up due to loss of inventory is expected to be low. The staff asked for some information on that during the audit, and bounded by the MHA versus -- time versus temperature curve. So the massive salt spilled, at least preliminary, is fairly low to the total mass of the system. And a lot of the heat of the system is tied up in the graphite, so you would expect that the temperature increase due to the -- to the loss of salt is pretty low.

MR. HALNON: How does it get out into the environment? It just infiltrates through the building, or is there --

MR. SCHMIDT: Yeah. It just -- it spills into an apartment or building and that just --

MR. HALNON: Is there any difference if the RBHKC continues to operate and sucks it out and pushes it out through --

MR. SCHMIDT: No.

MR. HALNON: -- point?

MR. SCHMIDT: No. We didn't look at that. This just goes -- just goes into the reactor building and out, part of that process, but --

MR. HALNON: Okay. Is that not a concern, then, that it could be funneled and dragged out by an operating fan and pushed out into -- with some velocity?

MR. SCHMIDT: Yeah. That would -- I guess that would have to be looked at as part of that. Its failure -- I mean, that's a control system that would lead potentially to a worse answer. But right now these are more, I would think, qualitative evaluation and not to that level of detail.

MR. HALNON: Okay.

MR. SCHMIDT: Methodologies for break air salt generation and Flibe, vessel-free surface evaporation. Methodologies are from the approved mechanistic source term topical report. Salt spill uses lower event-specific temperatures and, hence, lower fuel wetted graphite surface, tritium, and lower Flibe vessel preservice temperatures. That's just basically saying that the MHA temperatures are bounding this event.

Staff has reasonable assurance that the MHA would bound a salt spill based on the minimal heat-up in the low salt mass spilled. Quantitative dose assessment comparison between the salt spill and the MHA will be performed as part of the OL application.

Next slide?

The next event is loss of poor circulation. This, as Tim pointed out, is seizure of the primary salt pump, reactor trips on high outlook temperature, uncertainties as -- with most of these accidents will be quantified as part of the OL application.

Again, temperatures stay below the assumed MHA, hypothetical time or temperature versus time curve, except for the maximum reflector temperature and upper plenum temperature, which slightly exceed the free surface and graphite temperature limits for a short period of time. The same argument goes again.

The staff looked at some of the conservatisms that are in these calculations, and there are some significant conservatisms in these calculations that could be refined such that, you know, there is at least reasonable assurance that some of these values could be brought down. But we'll -- that will be determined as part of the OL.

Staffing scoping analysis, again, we did this event as well. It yielded similar results, as Andy will go through in the following slides. Staff has reasonable assurance the MHA does balance that -- balance that of the loss of poor circulation. Again, this isn't really a different release path than the MHA with effectively lower temperatures.

Next slide?

The pebble handling and storage system event, as was described as a break in the pebble handling system, it does have different release pathways. Reactor protection system trips to stop the pebble movement, as was described. Pebbles spill onto the transfer room, and no active heat removal is credited to limit the spilled pebbles temperature.

And I believe in the -- either the last figure or the second-to-last figure in APTR-018 has what the temperatures are for the pebbles. So that is available. And I can't remember -- it's one of the last figures in their illustrative examples for salt spill.

Release pathways, different from the MHAs. We've talked about this is basically the mobilized graphite dust that could come out as part -- that accumulated in the pebble handling system and then is expelled from the break, and then the pebble oxidation -- as Dr. Kirchner was talking about, there's assumption of spilled pebbles, and then any pebbles that remain in the pebble handling system that may be exposed to air.

We've had significant discussion with them to include -- make sure that all of those pebbles were included in the analysis, or will be included in the analysis, I should say.

MEMBER REMPE: So I see the temperature curve. Thank you. And it starts at xxx, and it just drops down. So I'm guessing they don't consider exothermic reactions if they start at xxx and they have --

MR. SCHMIDT: Yeah. I think it stated regime 1, if I remember correctly.

MEMBER REMPE: But I would think that you would have some exothermic reactions.

MR. SCHMIDT: It's very -- the oxidation at those temperatures is fairly low.

CHAIR PETTI: Yeah. But notice -- this is proprietary. We have to be careful.

MEMBER REMPE: Yeah. But I didn't say a number. I just said it's going down. I mean, it's

CHAIR PETTI: No. You did mention a number. You didn't mention what temperature scale, so you're okay.

MEMBER REMPE: Okay.

CHAIR PETTI: But if you notice that temperature scale, that's very low.

MEMBER REMPE: I've got documents that say anytime you're above 500C that you can have oxidation.

CHAIR PETTI: Oh, you can -- oh, for sure you can have oxidation.

MEMBER REMPE: Yeah.

CHAIR PETTI: But it's --

MEMBER REMPE: Yeah.

CHAIR PETTI: -- how much.

MEMBER REMPE: How much, but it can be exothermic, too, is what I --

CHAIR PETTI: Well, it's always exothermic.

MEMBER REMPE: Right. So then does this fit your --

CHAIR PETTI: But there's a huge amount of

MEMBER REMPE: Yes. Again, we --

CHAIR PETTI: You'll see --

MEMBER REMPE: -- do you know if their models considered the --

MR. SCHMIDT: There is an oxidation correlation that's used, and I did look at it. I'm not sure I remember it off the top of my head, but, yeah, there is an oxidation model. Yeah.

MEMBER REMPE: And it considers the --

MR. SCHMIDT: It was an oxidation model based on -- the Chinese had done a pebble matrix. They created an A3-3-type pebble, and they had developed a correlation that Kairos is referencing.

CHAIR PETTI: The U.S. has also done measurements of matrix material. It's in the literature.

MR. SCHMIDT: I was just referring to the ones that they referenced.

CHAIR PETTI: Yeah.

MR. SCHMIDT: It seemed like an appropriate reference over the appropriate temperature.

MEMBER REMPE: And so they are considering the heat input from that oxidation?

MR. SCHMIDT: The correlation is developed on basically mass loss. So whatever happens happens.

MEMBER REMPE: Okay. And then what about the reflector surfaces, too, within a system and that oxidation?

MR. SCHMIDT: That I think is part of the graphite topical report, and the oxidation rate of the graphite material is different than the ones I'm referring to for the pebbles.

MEMBER REMPE: Okay. So, anyway, it's just something that I thought --

MR. SCHMIDT: It's picked up in the graphite --

MEMBER REMPE: -- and that -- again, the answer may be there is not much combustible gas generated, but I just --

MR. SCHMIDT: Yeah.

MEMBER REMPE: -- didn't see those words anymore.

MR. SCHMIDT: You know, on this break, you know, I don't -- I don't personally have a good handle on how much structural graphite is exposed in this type of --

CHAIR PETTI: Well, you should -- you should look two figures earlier. There is the actual oxidation.

MEMBER REMPE: Okay. I see that.

CHAIR PETTI: So they are actually doing that.

MEMBER REMPE: And is this for the pebbles, or is this for -- so I'd have to go back and

CHAIR PETTI: This is for this accident, pebble handling.

MEMBER REMPE: -- pebbles. But this isn't the --

CHAIR PETTI: This could be for the pebbles.

MEMBER REMPE: -- reflectors, though. This is just the --

CHAIR PETTI: This is for pebble handling.

MEMBER REMPE: Just -- okay. But there is also --

CHAIR PETTI: In the pebble handling event

MEMBER REMPE: Okay.

CHAIR PETTI: -- the pebbles that spill on the floor.

MEMBER REMPE: Okay.

MR. SCHMIDT: Yeah. So, and the spilled pebbles are assumed to be at their maximum burn up, and, hence, maximum material at risk for the oxidation calculation, and then the dust activation uses the same assumptions.

Next slide?

Is that -- oh, yeah. I'm just basically saying I reviewed the pebble matrix oxidation and dust generation calculations, or methodologies to be more appropriate. The methodologies, I get those.

Fuel qualification topical report, so this is an important tieback to the fuel qualification topical report. You know, they're going to do tests for their own specific pebble matrix material, and that will inform how these calculations are done as part of the OL, right?

So right now they're using this surrogate A3-A that the Chinese had developed, but they're going to do their own testing to come up with their own, to see if that correlation is either still valid or needs a different correlation.

And, again, another tieback to the fuel qualification topical report, pebble wear will also be looked at, right? There's an assumption of the wear rate of these pebbles to generate that dust, right, that's expelled as part of the pebble handling. And I'm just referring back to they are doing tests to try to, you know, quantify that dust generation rate.

And, again, the dust generation

resuspension from the break is already discussed and approved in the mechanistic source from topical report.

PHS event uses lower temperatures. Again, I don't expect this to be really a temperature-driven event. The loss of mass is expected to be low. The salt, hence lower fuel wetted graphite surface temperatures with lower tritium releases, lower Flibe, vessel-free surface releases.

So, again, the concept is that the MHA temperatures will easily bound the PHSS, but you have to pick up these other figures of merit that have to do with dust and oxidation.

A quantitative dose comparison between the PHS event and the MHA will be performed at the OL application. These will be specifically compared at the OL application -- as part of the OL application.

Next slide?

This is a fairly simple thing that Matt was discussing from Kairos. So this is a radioactive release from a subsystem or component. The short answer is that the materials at risk have to be limited such that if there was, say, a single event, say, speculated seismic event, that the non-protected structures or non-safety-related structures, I should say, will release, but that will still be bounded by the MHA based on the quantities of material at risk in these areas.

Next slide?

So general challenges to normal operation, as was discussed, are caused by inadvertent operator action, failure of a control system or instrumentation. The reactor protection system will sense to terminate the event, assuming setpoints are reached. Events caused by operator action, control system, instrument failures, are typically bounded by events analyzed in Chapter 13 due to the use of bounding assumptions and analyses.

Consequences caused by inadvertent operator action, control system, or instrument failure will be reviewed in more detail as part of the OL application.

Next slide?

Internal or external events. Again, these are -- typically limiting internal events are primarily just by Chapter 13. Kind of an aside to that is the fire protection, which isn't really addressed by Chapter 13. Programs are addressed as part of PSAR Section 9.4 and will protect safety-related systems that perform event mitigation function.

Most external events are addressed by designing SSCs commensurate with the hazard of the applicable standard. Seismic-induced reactivity event due to unique -- that's unique to the pebble bed, this was kind of a -- this is a different, you know, technology-specific accident that's, you know, driven by an external hazard being a seismic event.

So there was -- Kairos did look at some of the increase in pebble -- pebble packing fraction, sorry, and associated reactivity increase. As we'll discuss probably in the excess reactivity, this will be I think easily bounded by the insertion of excess reactivity event.

They did look at the change in moderation near the reflector where it's a positive reactivity, and then a corresponding negative reactivity insertion towards the middle of the pebble bed. No final numbers were generated, but there is a release reported in the -- in the technical report.

But -- so there is a plus and a minus component associated with this packing fraction increase. I did a little research as far as relative to high-temperature gas reactors, especially the Chinese -- I think it's H-10, HT-10.

CHAIR PETTI: HRT-10.

MR. SCHMIDT: HTR-10. Thank you. And one of the -- you know, one of the reactivity increases that you don't expect to see in this type of positive buoyancy bed is slumping of the core relative to, say, control rod insertion. Right? So if you were to repack this thing, you would expect that since it's positively buoyant to actually pack towards the top of the core and not slumped towards the bottom.

So you're going to be moving the core effectively in the direction of the control rods. You know, there's a pretty big reactivity insertion potentially, depending on where your rods are inserted in a high-temperature gas reactor because the pebble bed will slump on an increase in peaking factor, and you'll effectively have less rod insertion as part of that.

So there's like a two-part reactivity insertion, one due to the slumping, due to the increased packing fraction.

So that -- that situation should not occur in the Kairos design. Therefore, I expect that the excess reactivity event, which we'll talk about in detail when we get to the following slides, will bound this basic --

MEMBER MARCH-LEUBA: Do you have any idea what the packing fraction is? Was it the maximum theoretical -- is it 0.1 percent, or is it 10 percent?

MR. SCHMIDT: Do you mean as far as --

MEMBER MARCH-LEUBA: With respect to the with respect to the maximum theoretical you can put the bolts on?

MR. SCHMIDT: So I want to say it's like 60 percent of .6 is -- is the number that I'm recalling. But I'm not 100 percent sure on that.

MEMBER KIRCHNER: That's about right, Jeff. This is Walt. Yeah.

MR. SCHMIDT: Okay.

MEMBER KIRCHNER: For a static pebble bed reactor, that's about it. It depends also on the diameter, because you have --

MR. SCHMIDT: Right.

MEMBER KIRCHNER: -- the edge effects on the density of pebbles.

MEMBER MARCH-LEUBA: I'm not asking about how much space there is for the Flibe. I'm saying what you are talking about actually when you shake it during the --

MR. SCHMIDT: Oh. How much --

MEMBER MARCH-LEUBA: -- and it compresses, are you going to get more?

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: So what this -- with respect to the maximum theoretical you could have spheres.

MR. SCHMIDT: I don't remember that number. I think it was actually in the article I read for a high-temperature gas reactor, but I don't recall it. And I don't know if I thought it was --

MEMBER MARCH-LEUBA: Because you --

MR. SCHMIDT: -- overly applicable to this.

MEMBER MARCH-LEUBA: -- you need the number to know what the --

MR. SCHMIDT: Yeah, yeah. You do. You do. You're right. You know, this -- like I said, I expect the bed to actually move up, and it will densify to some amount.

MEMBER MARCH-LEUBA: Because I --

MR. SCHMIDT: Yeah. Due to the shaking.

CHAIR PETTI: If it's the paper I think you read, because there aren't that many out there --

MR. SCHMIDT: Yeah. No, it was hard to find.

CHAIR PETTI: -- it was done by people I know. I think they went -- they assumed it went to maximum packing, which Ron says is .72.

MR. SCHMIDT: Yeah. So that does seem -- that sounds familiar, the .72. But I'm --

MEMBER MARCH-LEUBA: .72 is the maximum packing. So what is the normal operating --

MR. SCHMIDT: .6 roughly I think is --

MEMBER MARCH-LEUBA: Okay. So --

MEMBER BALLINGER: Basically, it's .74.

MR. SCHMIDT: .74, okay.

MEMBER MARCH-LEUBA: You just calculated it?

(Off mic comment.)

MEMBER MARCH-LEUBA: So you calculated from .6 to .7, so that's -- that's not the packing.

MR. SCHMIDT: Yeah. Again, I think we're going to have -- my last bullet there is we're going to have to look at this in detail at the OL. So I think this will be one thing that will be revisited. I was just looking for information that I could use for a reasonable assurance finding that excess reactivity would bound this event.

MEMBER MARCH-LEUBA: Just go with the binding, the earthquakes takes it to the maximum theoretical.

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: And you just need to know what the normal is.

MR. SCHMIDT: Right. Right. But there is -- yeah, right. You could do that, but there is actually a negative reactivity insertion --

MEMBER MARCH-LEUBA: Into the control --

MR. SCHMIDT: -- due to Flibe -- well, the Flibe -- let's just also -- the bed moving up relative to the control rods, but you could also assume that the controls rods are not -- that you're fully withdrawn.

MEMBER MARCH-LEUBA: That's a good, handy theoretical approach.

MR. SCHMIDT: Yeah. I think we'll address that as part of the OL. How about that?

All right. Next slide, please.

Okay. So this is an area of -- so it's prevented events, so these are events that are not analyzed as part of the PSAR, and I'm going to -- there is a list in this PSAR Section 13.1.10. I'm not going to go -- I didn't -- I'm not going to go through all of the prevented events, but I will highlight two that I thought were the most significant that the staff passed RAIs on.

The first one was RAI-348, asks the basis of why recriticality or unprotected events are excluded from consideration. Kairos modified -- in response to that RAI, Kairos modified PSAR Section 4.2.2.3 to further describe the shutdown element testing to ensure the shutdown margin analysis remains valid, and in part lower the probability of an unprotected event.

So, as you recall from our previous discussions, the shutdown rods go into the pebble bed. So the staff was concerned that -- didn't have a lot of experience, the insertion of rods into the pebble bed and that they would sufficiently go in to both meet the shutdown margin assumption and actually go into the core enough to prevent the unprotected event.

So staff asked that -- what type of qualification testing was going to be performed to ensure that those two items were met, and Kairos modified the PSAR section to address that.

The main thing the staff wanted to get out of that is to ensure that if you were to insert all of the control rods, would they successfully go into the pebble bed to a sufficient depth to ensure shutdown margin and prevent recriticality, because, you know, as you cool down, right, you're going to add positive reactivity to the system again, and you have to have enough excess reactivity to maintain shutdown. And the other one was just to ensure that you didn't have an unprotected event from a mechanical-induced mechanism.

And then RAI-350 asked in part how component integrity is ensured for the duration of an air ingress event, including air ingress beyond the heat rejection blower trip, and that was addressed in SE Section 5.1.3.2.6, addresses the material qualification testing after seven days.

And then there was discussion beyond what happened -- what happens beyond seven days, and could this system be placed in a safe state, because the air for the air ingress event could -- might proceed beyond seven days.

In the discussions with the Applicant, the staff reached reasonable assurance finding that the reactor could be placed in a safe state, protect public health and safety.

And so now I'm going to turn it over to Andy Bielen in Research, and he is going to go through some of the scoping analysis.

MR. BIELEN: Hello? Can you hear me?

MR. SCHMIDT: Yeah, we can hear you.

MR. BIELEN: Okay. I'm going to make you look at my face, because I did put on a jacket.

(Laughter.)

MR. BIELEN: Okay. So, yes, I'm Andy Bielen. I'm in the Fuel and Source Term Branch of the Office of Nuclear Regulatory Research. At DANU's request we performed a series of scoping calculations for the PSAR review.

So first I want to remind you that as part of our non-LWR RADIS plan we have been over the last several years doing some public demonstrations and workshops of our ability to simulate the relevant phenomena and characteristics of non-LWR systems. Specifically, Volume 3 covers severe accidents and source term analyses.

Within that suite of models that we've developed is included the UC Berkeley Mark 1 design which represents TRISO Pebble Fuel Molten Salt Cooled FHR technology. Oak Ridge National Laboratory uses scale suite to generate inventory and reactor physics data, among other things, which is then provided to the MELCOR severe accident source term code that Sandia develops so we could model different accident progressions.

Next slide, please. So it's nice that we did these workshops for the past few years because when DANU actually had an application in hand, they asked us if we could provide any support, and we were able to really go in and do some modifications to our existing models to make it look more like Hermes and then run some analyses that I think they found to be useful in informing their engineering judgment.

One of the things I want to kind of point out here is, as I mentioned, the original demonstration workshops were very much in the severe accident source term regime. We focused on the UCB Mark 1 design as we understood it. We focused on fission product release from the TRISO and into the buildings and all these other sorts of things.

The focus was on beyond-design basis events. We were explicitly doing elemental tracking, radioisotopes and that sort of thing to figure out if something went very, very, very wrong, where would all this stuff end up.

In contrast to that, with respect to Hermes, we were asked to do this. We were asked to basically provide an independent verification of some of the specific event evaluations that Kairos had presented to ensure that the temperature stayed within the MHA envelope that they've been describing over the last few hours.

We wanted to do this with a very quick

turnaround to support the licensing schedule. We wanted to keep this as quick and transparent as possible. All of our information in the models was informed by the PSAR information that was readily available, or in the absence of specifics, engineering judgment to the best that we could.

As I said, and as described in our meeting back in March, we use scale to generate inventory, decay heat, power shapes, and all these sorts of things. Then we analyzed two classes of transients from the Safety Analysis Technical Report. Specifically, the insertion of excess radioactivity scenario, and then a couple flavors of loss of for circulation.

Okay. So to kind of walk through the MELCOR modeling approach. So as you know, and you've heard many, many times over the course of these meetings, we are very much in preliminary space here so we don't have a whole lot of detail design information available to us.

We have focused our modeling efforts on what we know in the primary system. The intermediate loop and the DHRS are both represented basically by boundary conditions at this point. We just don't have any better information to build models based off of.

In fact, some of the flow geometry and structures at the top of the core specifically. I don't think we have enough detailed information to really know how everything is specifically arranged, but we think we know enough to generate models that can come to meaningful conclusions.

The pebble bed itself is modeled via porous media approach. We have made the geometry and the nodalization between the scale models and the MELCOR models be consistent in order to simplify the mapping process.

The reflector itself, I'll say that I think it was judged that we just didn't have enough information about what the flow splits looked like, what was bypass, what was active core, so we just neglected to model bypass at this stage. I think that would be something that we would definitely revisit when more detail was available.

Is there anything else I wanted to make sure to mention at this point? I think that's pretty much it. Again, the reflector, I think, that's another thing where we don't have a whole lot of specifics on what this thing looks like yet. It was modeled approximately within the uncertainties that we -- within the information we had available, but we would certainly like to sharpen out pencils as far as that goes.

One more point. Since the fluidic diodes were relied upon to provide a natural circulation of flow path under accident scenarios, we do explicitly model those. You can kind of see that flow path there at the top of the model underneath the primary salt pump. The model is basically very simple. Kind of check valve almost with a very high loss coefficient in one direction and a very low one in the other direction.

Okay. Specifically talking to the DHRS, so the whole goal here is to basically be able to model effectively the heat transfer from the core out to the ultimate eat removal system in as much detail as we need to. We start in the core and we work out way through all the layers, through the pebble bed to the reflector, through the reflector out through the downcomer to the core vessel.

Then from the reactor vessel we allow radiation and convection within that compartment to transfer eat into these DHRS thimbles basically. Then the DHRS model, you know, basically we have a boundary condition that looks like 100 degree C model, infinitely replenishable 100 degree C boiling water.

Right? So when we want to model degraded conditions within the DHRS itself, we can do that by basically turning on and off heat transfer surfaces based on the number of trains that we want to evaluate.

Yeah, there are certain parameters within this analysis, like when you're talking about radiation heat transfer you have to worry about emissivity and that's something that we have available to us to do sensitivity analysis or calculations with. Convective heat transfers is something else that we have looked into. Then the specifics of the thermal resistance within the DHRS itself.

Then just to kind of point out that we basically took the Hermes system and plopped it into the UCD1 building, right? We know that's not what the real thing is going to look like. There's a lot of kind of uncertainty or approximations made within the specific dueling geometry itself, which is another reason why we didn't go forth and do like specific source term calculations because, you know, we know the real thing is going to be different.

Okay. Next slide. Okay. So before I get into describing the specifics of these simulations, I wanted to point out first that these are simulating basically three days of simulation time. Those required about 10 hours of execution time to produce these curves.

The MHA -- the temperature curves that are provided by the MHA analysis are the solid lines on this graph. You'll see the green solid line is what the fuel temperature is allowed to get to. The red solid line is what the stainless steel structures are allowed to get to. The purple solid line is what the reflector or the graphite structures are allowed to experience.

Then the blue solid line is the flag freezing temperature. The whole idea of this approach is as long as your deterministic evaluation lies within this envelope, then you can say that you have met your dose requirements. And so our MELCOR models have a couple different flavors of hot pebble, if you will.

When you generate a peak fuel temperature plot, you have to find some way to make like -- to represent what the hottest part of the core is including all the uncertainties that you want to put on that hottest part of the core so you have operational flexibility. We have basically two flavors of hot pebbles in our model.

The first one, the very fine dashes here, you can see those are basically -- as far as the MELCOR model is concerned, those are a pebble with TRISO particles that have been bumped up to the very top edge of the allowable power envelope from the AGR sequence of tests.

Then we have another -- we noticed when we ran the initial set of calculations that, hey, we don't match initial peak temperature very well with the applicant so we have another version of a hot pebble where we just like turned the power up on that pebble until we got something that was reasonably comparable.

I think, you know, in retrospect maybe we should have looked at some of the sensitivity co-efficients on the different heat transfer models that we have available to us in MELCOR and done some adjustments on that as well as power uncertainties. You know, suffice it to say that we have some treatment of this hot pebble in the MELCOR models.

Before I get into the specific results, does that seem -- you know, are there any questions at this point?

MEMBER REMPE: Sure. I have a question

just to make sure I'm understanding what you're saying. There's like a green dash line with very small dots and it only gets to about 1120C.

MR. BIELEN: Right.

MEMBER REMPE: And then you've got something where you just arbitrarily jacked up the power to 1380 or something like that? Is that what you're telling me? And it's still below the 1400 something or other limit?

MR. BIELEN: Yes. I don't know if I would use the word arbitrarily necessarily but, yes. Essentially what we've done is we -- so we're not doing any direct manipulation of the heat transfer models themselves. Right? So the knob we're turning is particle power.

The fine dashes are the -- or the dots basically are what happens if we have a hot pebble that bumps up the power with nominal heat transfer coefficients, although this pebble has been placed in the hottest location of the core.

Let me be clear about that. But what happens when we bump the particle power of that pebble up to the AGR limit? I don't remember specifically. It's like 255 milliwatts per pebble or -- I can't remember the specific number.

MEMBER REMPE: So you kind of picked a peak value that was in a representative range of values?

MR. BIELEN: Right. From the AGR test basically. So we know that we have an envelope that lives there and we can go and push a single pebble up there. This is what the temperature looked like. Now, clearly when you look at the applicant's analysis, they have done some other manipulations that I think are, you know, under the proprietary wall that are getting their peak temperatures even higher than that.

In lieu of going in and manipulating our heat transfer mechanisms, what we've done is basically just, yes, we have tuned the power of the peak pebble to try to get a temperature that looks like what the applicant has produced.

MEMBER REMPE: Okay. And I'm guessing you don't have enough information yet to really see what parameters are really important. For the future when the real design comes in and you try and model it more with the actual design details, do you know yet, you know, this parameter is going to be really important rather than the peak power of the pebble?

MR. BIELEN: Right, yeah. And I think, you know, that is a good question. I think with robust --

a little bit more robust kind of long-term planning of the analysis we would provide to support DANU in this particular case, we would be prepared to perform some sensitivities up front and say, okay, well -- and look at hot channel methodologies that are out there.

Ideally we would have access -- our model developers would have access to the hot channel methodology that Kairos is using and being able to specifically adjust the different aspects of this heat transfer that they are adjusting and see if we get kind of simpatico affects on how your figures of merit change as you change your model parameters.

MEMBER REMPE: Thank you.

MR. BIELEN: Sure. Okay.

Yes.

MEMBER KIRCHNER: This is Walt Kirchner. Just one quick question. Did you assume one of the DHSR trains down for this particular plot?

MR. BIELEN: I think the base case was one DHSR train unavailable of the four.

K.C., you can step in if that's wrong.

MR. WAGNER: I think that's what the applicant used, too.

APPLICANT: Yes, that's correct. We used three.

MR. BIELEN: Okay. Right. The analysis that I'm going to present to you here is just very base case as close to the technical report as we could generate. There are a lot of additional work that we did behind the scenes to kind of get a feel for the importance of various systems' availabilities and other parameters, but we can't really discuss that here unfortunately.

So, okay. In terms of the reactivity insertion event, as Kairos kind of discussed in their part of the presentation here, you're reporting a lot of reactivity and relatively quickly. Three dollars, you know, in LWR space is like impossible and, you know, not a thing that can even physically happen. Three dollars in 100 seconds is a lot.

But basically 10 seconds into the withdrawal, you end up tripping out on high power. As an additional conservatism here they have a primary salt pump trip and a flow coast down. You see here we do get a fuel temperature increase initially due to that power increase, which is pretty quickly stamped out by the trip.

You can see the latent effects of that heat leaving the fuel and getting into other parts of the system as the transient progresses. Then you just sort of sit there for a long time until the scenario is terminated.

Again, we think that our results -- you know, given all the uncertainty we have in the specifics of the sand models versus the MO core models, we think our agreement is, you know, pretty reasonable. We feel fairly comfortable that what Kairos has presented is reasonable.

I think I forgot to mention this, but the reference results we're using are that little box on the upper right. The PSAR results are the little box on the upper right. The MELCOR calculations are the big box.

Okay. Next slide. As Kairos has eluded to, they have two flavors of loss of for circulation; one for overheating trying to maximize temperatures, and then one for over-cooling to try to see if they can freeze the flood.

We looked at both those scenarios I'm going to present here on this slide what the MELCOR results were for the overheating scenario. You have a primary salt pump trip that is actually a seizure so you have a very rapid decrease in flow rate. You end up with a trip over temperature. The temperature is coming up during the transient.

Then again, as soon as the reactor trips, you have some kind of latent heat that leads the fuel. Over the course of time it's transferred out through all the heat transfer pathways in the system to the DHRS.

Eventually your heat removal exceeds your heat generation and then you start coming down in temperature after about a day or so, or a little after a day. Again, a case where you're clearly within DMHA envelope. As the DMHA envelope is appropriately defined, these transient scenarios would be pretty, you know, within the acceptance criteria.

MR. SCHULTZ: Andrew, this is Steve Schultz. The relative comparison between the results that you've obtained and those that Kairos has developed is encouraging thinking about moving forward to the operating license analyses. Didn't you feel that?

MR. BIELEN: Oh, yes. I mean, you know, we were working on this project in close collaboration with DANU. They are under a lot of time pressure to get these reviews done quickly and efficiently.

I think Jeff can speak to this himself but, you know, my impression throughout the whole course of the project has been, hey, you know, by virtue of you guys doing these things and us seeing some very promising kind of agreement between different completely independent methods doing the same sorts of simulations.

It's a lot easier to say, okay, I have some comfort here. We know that we're going to need to do some more work when the OL comes in, but at least we have -- it definitely cushioned that ability to get to a reasonable assurance for a construction permit.

MR. SCHULTZ: That was impressive to me. I really appreciate you showing us the detail.

MR. SCHMIDT: This is Jeff. I just want to say my two piece here. I was amazed the general trends of the curves were so similar. That was --

MEMBER MARCH-LEUBA: Also submission of energy.

MR. SCHMIDT: Yeah, right, but there are ways to screw that up, as you well know, Jose. I don't know. I --

MR. BIELEN: Jose has never messed anything up.

MR. SCHMIDT: Yes. I was encouraged by the results; the shape of the curves, the times to trip, the general trends of the curves. The fact that, you know, even when we were pushing particle powers to the edge of the AGR envelope, we still had a lower temperature than the applicant.

That's kind of what I was referring to. It's part of the audit that there were conservatism in the applicant's calculations. This clearly helped highlight some of those.

I can get some comfort in the fact that there were conservatism. I generally was very impressed with the likeness of the results based on the information that we had available and the time that researchers in Sandia had to do this.

MEMBER MARCH-LEUBA: I've seen the slide you mentioned of the cooling and freezing. Could you give us some thoughts on that?

MR. SCHMIDT: You know, the over-cooling analysis in the PSAR is from the loss of poor circulation and with four trains. We didn't put -- while we did it for comparison, we don't necessarily think it's the limiting condition.

MEMBER MARCH-LEUBA: The important thing is if it leads to a pathway for reuse, which is different. I don't know.

MR. SCHMIDT: Right now the working assumption is that freezing will be prevented. I just wanted to point out that, you know, I didn't spend much time talking about the over-cooling analysis in the PSAR because I don't necessarily think it's the limiting over-cooling event. While it is informative, I think there could be other situations that may be more limiting and that will have to just be flushed out as part of the OL.

MEMBER MARCH-LEUBA: The thing is with reactors we have to worry about a number of events. Clear thing that we flag is freezing. You have to use some thought and make sure that doesn't produce any unexpected events.

MR. SCHMIDT: Yeah, as we discussed in the decay heat removal, that is clearly on the mind of the staff of like what scenarios after you were to say we are to activate the system that you could get to, say, a freeze within 72 hours.

MEMBER MARCH-LEUBA: Is the bundling condition 100 degrees C?

MR. SCHMIDT: Yeah, yeah. Right.

MEMBER MARCH-LEUBA: On the vessel?

MR. SCHMIDT: Well, on the decay heat removal system, yeah. Right.

MEMBER MARCH-LEUBA: And the freezing is --

MR. SCHMIDT: 450.

MEMBER MARCH-LEUBA: 450?

MR. SCHMIDT: 459 C, I think.

MEMBER REMPE: When you finally get the OL are you planning to do some sort of confirmatory dose calculations? I mean, right now what I think I'm reading is we're going to use the MHA and that's all we're going to do for a dose calculation. Then we'll do analyses and compare it to metrics. These are being compared to those metrics. They are not dose calculations. Are you going to --

MEMBER MARCH-LEUBA: You will do these calculation versus the figure.

MEMBER REMPE: Right. Is that all staff is going to do, too? Are you going to do confirmatory on the MHA?

MR. SCHMIDT: I don't think it's been decided. We have not laid out a path in detail where we're going to go. This was just to inform our reasonable assurance finding for the construction permit.

MEMBER REMPE: Sure.

MR. SCHMIDT: And to reinforce what we thought our engineering judgement was. Beyond that, we're not committing to anything at this point other than we have the models and capability to do it.

MR. BIELEN: That's exactly right, Jeff. The only thing I'll say about that right now is, first of all, DANU is our customer and we aim to please our customer, but we have the models and capabilities. You know, I think a reasonable person would say, uh, if you can do this thing, why don't you? That's all I'll say about that at this point.

MEMBER REMPE: So then I'm going to mention to you then you've got the capabilities in MELCOR to look at oxidation of the pebbles and the reflector graphic. I think you probably also have the ability to predict Co and Co2 forms. Is that true? I'm not sure actually. I shouldn't say I think. I don't know what all models you put in for gas reactors.

MR. BIELEN: Yeah, we'll have to defer to K.C. on this.

MR. WAGNER: Yes, we have empirical correlation and the ratio of Co versus Co2's function of temperature.

MEMBER REMPE: So you could do that type of calculation, too.

MR. WAGNER: Yep.

MEMBER REMPE: Thank you.

MR. SCHMIDT: This is Jeff Schmidt. Do you have more that you want to go through or are you done?

MR. BIELEN: I mean, I'm prepared to respond to any additional questions, but I think that's my last slide.

MR. SCHMIDT: Let's go on to the next slide then. I'm going to do this one. Overall staff conclusions. The staff found that postulated event methodologies can be used to predict conservative event temperatures and dose releases. This is really the calculational framework of some of the things I talked about like dust generation, associated activities associated with that dust generation, oxidation and Oxidation correlations.

I has, I think, a very reasonable framework. Staff reviewed PSAR Table 13.1-1, Acceptance Criteria, and found these acceptable as described in SE Section 13.2.2 because they account for the physical phenomena and release pathways that are not part of the MHA to ensure that the MHA remains founding.

The OL application will provide dose analysis for events honored by the MHA release, along with the comparison to the acceptance criteria for the figures of merit in PSAR Table 13.1-1.

Next slide. Because the figures of merit and associated acceptance criteria ensure that the MHA releases remain amnii, the staff has reasonable assurance that the radiological consequences of the postulated events will also meet regulatory requirements of 10 CFR 100.11, and 10 CFR 50.34(a).

Staff concludes information in the Hermes PSAR Chapter 13 is sufficient for the issuance of a construction permit (CP) in accordance with 10 CFR 50.35 and 50.40. Further information can be reasonably left to the OL application.

MEMBER REMPE: I'm sorry. I guess I misunderstood. The third bullet, they will provide dose analyses for events -- for each category events even though it's down by MHA. I thought they said no, we're just going to do the MHA and --

MR. SCHMIDT: Yeah. So my expectation is like specific classes of events the limiting of that will be compared to the MHA.

MEMBER REMPE: Will be compared with the metrics. Here it says they will provide dose analysis.

MR. SCHMIDT: Dose analysis.

MEMBER REMPE: Someone asked that today and I thought they came back and said no, we're just going to do the MHA.

MEMBER MARCH-LEUBA: My understanding of the answer was that they wouldn't.

MEMBER REMPE: That's what I thought, too, but you are confident what you have here is true.

MR. SCHMIDT: As far as I'm aware, yes.

MEMBER REMPE: Is that documented enough in the SE that we can be confident? That's why I was pushing for the staff to do the dose analysis if they are not going to.

MR. SCHMIDT: It's not our responsibility to do the dose analysis.

MEMBER REMPE: Yeah, I know, but --

MR. SCHMIDT: I think it basically says in the SE that they -- I would have to go back.

MEMBER REMPE: Let's ask the applicant again but Jose came away with the same response.

MEMBER MARCH-LEUBA: SE is bounding. You can put an additional condition but --

MR. SCHMIDT: The SE does not --

MEMBER MARCH-LEUBA: -- the oil --

MR. SCHMIDT: That's true. The SR -- the PSAR in this case dictates.

MEMBER MARCH-LEUBA: All the staff can do is wait for the applicant to make up their mind and then decide whether --

MEMBER REMPE: So we don't necessarily believe this third bullet. I think I heard you say I'm not sure.

MR. SCHMIDT: I mean, that's my expectation.

MEMBER REMPE: Is that expectation documented in your SE?

MR. SCHMIDT: That I would have to go back and see if it's clearly documented.

MEMBER REMPE: can we just ask the applicant to clarify because Jose and I kind of came away with a different response.

MEMBER MARCH-LEUBA: Let's be realistic. The II process is a work in progress.

MR. SCHMIDT: I mean, we can ask our --

MEMBER REMPE: Jose, I thought he asked them that and they said something different.

MR. SCHMIDT: He did. He did. My bullet is likely different than the response earlier.

MEMBER REMPE: Okay. Do we want to clarify it or let it go?

MEMBER MARCH-LEUBA: It's clear that the applicant doesn't have to commit now to do anything.

MEMBER REMPE: Okay.

MEMBER MARCH-LEUBA: Just provide a --

MEMBER REMPE: Okay. We're both puzzled at the difference, I guess.

MR. SCHMIDT: Okay. I thought so.

MEMBER REMPE: I'm sorry, what?

MR. SCHMIDT: He was giving me guidance reminding me what's in our SE. We believe it's in our SE.

MEMBER REMPE: Good. Okay.

MEMBER KIRCHNER: This is Walt. I believe it's in your SE. It's my understanding that your last bullet is correct.

MEMBER REMPE: I would like to see what -- point me to the place. I've got the SE here and it would help.

MR. SCHMIDT: It's listed in the Appendix A.

MEMBER REMPE: So it's in Appendix A? That's great. Okay.

CHAIR PETTI: Is that it?

MR. SCHMIDT: It is.

CHAIR PETTI: So, members, any questions? With that, the presentations are done. I think at this point we probably should go to public comments and then we can talk about next steps after that.

MEMBER MARCH-LEUBA: The memo is -- the

discussion needs to be transcribed.

CHAIR PETTI: Right.

MEMBER MARCH-LEUBA: The public comments also.

CHAIR PETTI: Okay. Any member of the public that has a comment, please unmute yourself, identify who you are, and state your comments.

Okay. Not hearing any, I think we are done with presentations. We have to decide whether you would like -- we will go off the record, court reporter.

(Whereupon, the above-entitled matter went off the record at 4:20 p.m.)

Kairos Power

KP-NRC-2304-004

April 11, 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 Section 12.9 and Chapter 13

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 April 18-19, 2023 briefing for the Advisory Committee for Reactor Safeguards (ACRS), Kairos Power Subcommittee. During the April 18 meeting, participants will discuss Hermes Preliminary Safety Analysis Report (PSAR) Section 12.9 and Chapter 13. During the April 19 meeting, participants will have additional discussion on Hermes PSAR Chapter 13.

Enclosure 1 provides the non-proprietary slides for the April 18, 2023 briefing. Enclosure 2 provides the non-proprietary slides for the April 19, 2023 briefing. 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 Rachel Haigh at haigh@kairospower.com or (704) 412-5920, 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 April 18, 2023 ACRS Kairos Power Subcommittee Meeting (Non-Proprietary)
2) Presentation Slides for the April 19, 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 April 18, 2023 ACRS Kairos Power Subcommittee Meeting
(Non-Proprietary)

Kairos Power

Hermes PSAR 12.9 Quality Assurance

JORDAN HAGAMAN – DIRECTOR OF RELIABILITY ENGINEERING AND QUALITY ASSURANCE

ACRS KAIROS POWER SUBCOMMITTEE MEETING

APRIL 18, 2023

12.9 Quality Assurance

- 10 CFR 50.34 (a)(7) “A description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility.
- The Quality Assurance Program Description (QAPD) for the design, construction, and operation of the Hermes reactor is based on ANSI/ANS 15.8–1995 (R2005), “Quality Assurance Program Requirements for Research Reactors”
  - Endorsed by NRC Regulatory Guide 2.5, “Quality Assurance Program Requirements for Research and Test Reactors” (RG 2.5)

Quality Assurance Program Description

- The Hermes QAPD applies to design-phase, construction-phase, and operations-phase activities affecting the quality and performance of safety-related structures, systems, and components (SSCs).
- Safety-related SSCs within the scope of the Hermes QAPD are identified by design documents. Technical aspects are considered when determining program applicability including, as appropriate, the SSCs design safety function.

Quality Assurance Program Description

- The Hermes QAPD includes discussion of eighteen design, construction, and modifications program elements:
  - Organization
  - Quality Assurance Program
  - Design Control
  - Procurement Document Control
  - Procedures, Instructions, and Drawings
  - Document Control
  - Control of Purchased Items and Services
  - Identification and Control of Items
  - Control of Special Processes
  - Inspections
  - Test Control
  - Control of Measuring and Test Equipment
  - Handling, Storage, and Shipping
  - Inspection, Test, and Operating Status
  - Control of Non-Conforming Items and Services
  - Corrective Actions
  - Quality Records
  - Assessments

Hermes PSAR Chapter 13 Accident Analysis

DR. MATTHEW DENMAN – DISTINGUISHED ENGINEER, RELIABILITY

ACRS KAIROS POWER SUBCOMMITTEE MEETING

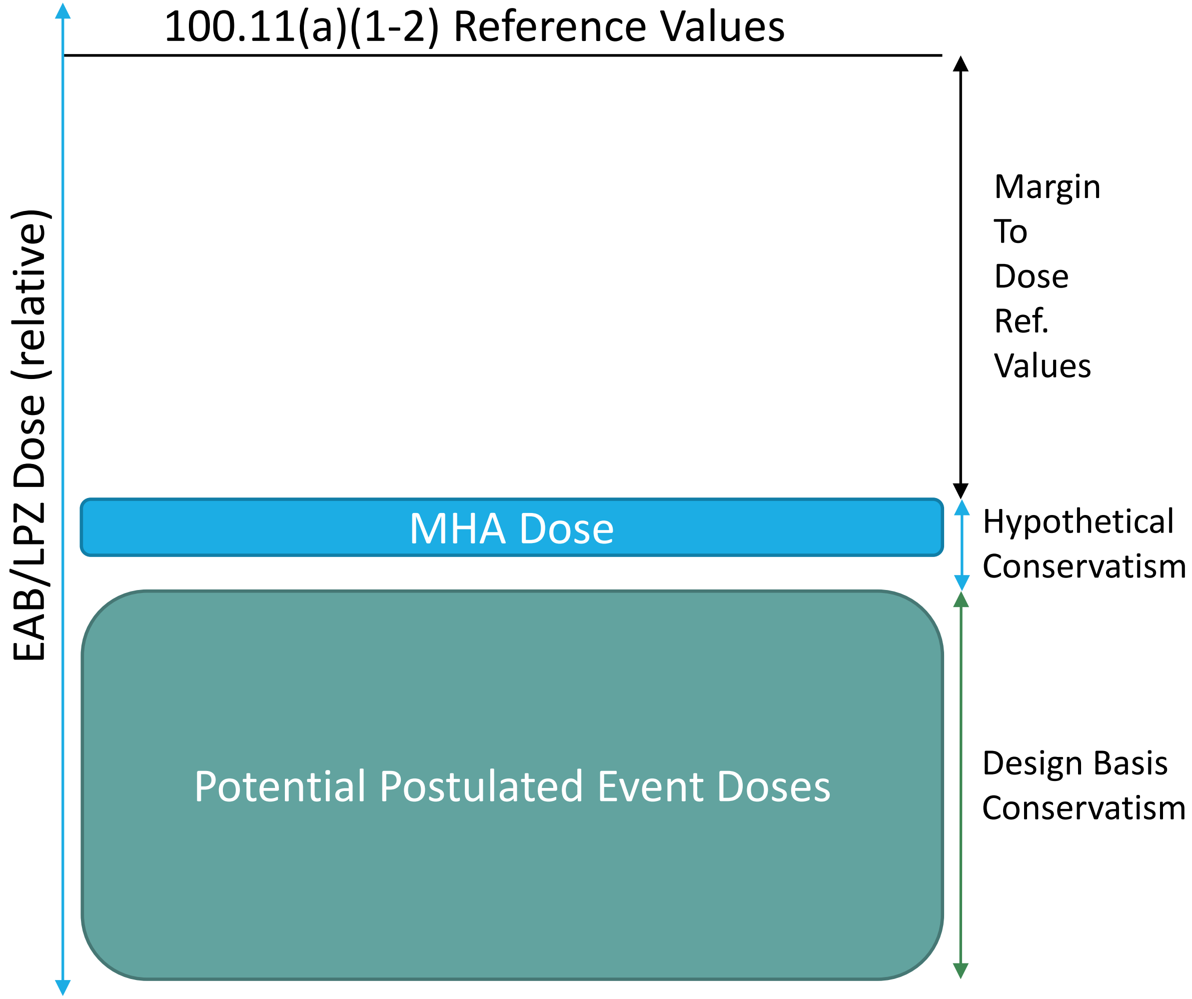
APRIL 18, 2023

Safety Case Summary

- 10 CFR 50.34(a)(4) requires a preliminary safety analysis to assess the risk to public health and safety from operation of the facility, including determination of the margins of safety
- To demonstrate compliance with 10 CFR 100.11 dose reference values, a Maximum Hypothetical Accident (MHA) that bounds the postulated events is analyzed for dose consequences by challenging the performance of functional containment
  - The Hermes MHA approach is consistent with guidance in NUREG-1537
  - The Hermes MHA is not physical
  - The Hermes MHA includes conservatisms that maximize source term
  - The Hermes MHA includes a postulated release of radionuclides
- To ensure that the postulated events are bounded by the MHA:
  - The list of postulated events is comprehensive to ensure that any event with the potential for significant radiological consequences has been considered
    - Initiating events and scenarios are grouped, so that a limiting case for each group can be qualitatively described in CPA (quantitative results will be provided with OLA)
    - Acceptance criteria are provided for the important figures of merit in each postulated event group to ensure that the potential consequences of that event group remain bounded by the MHA as the design progresses
  - Prevention of an event initiator is justified in the PSAR

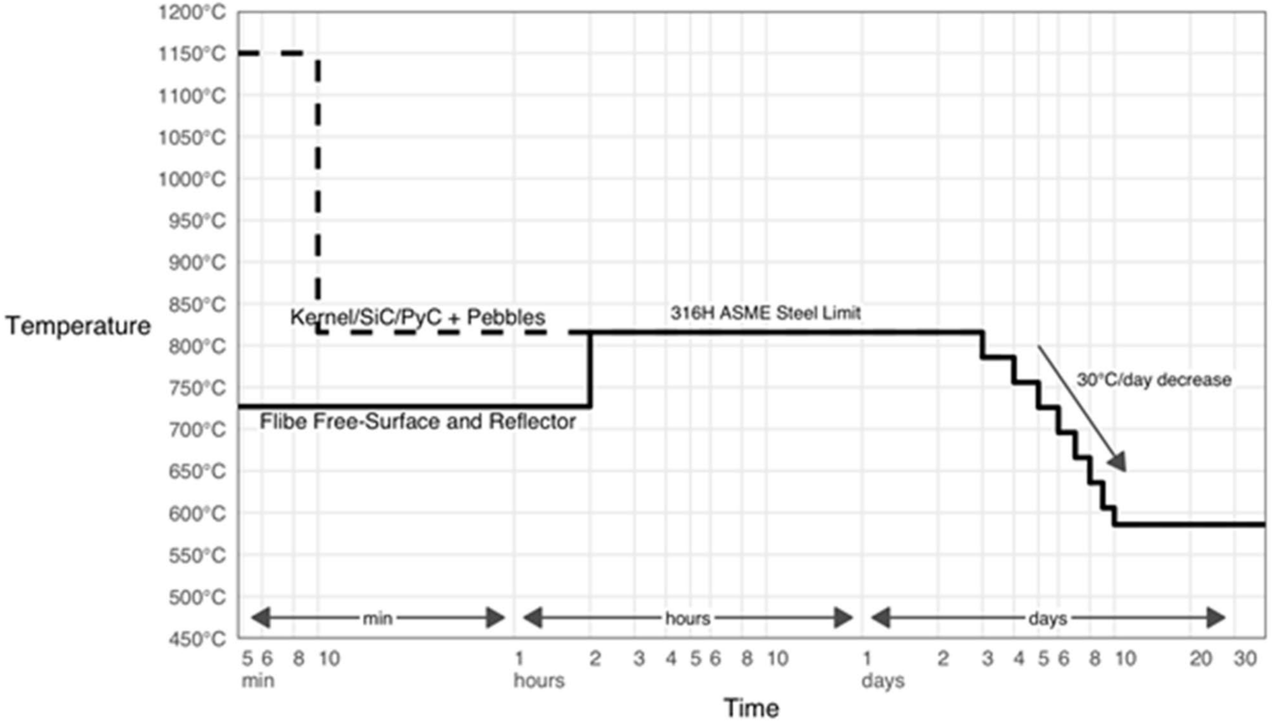
Relationship between Dose Limits, the Maximum Hypothetical Accident, and Postulated Events

- The Maximum Hypothetical Accident (MHA) is constructed to:
    - Be conservatively non-physical to overestimate potential off-site dose consequences
    - Provide confidence that sufficient safety margin exists
    - Ensure that reasonable design constraints will result in bounded postulated event doses
- At the PSAR stage, only the MHA dose is:
    - Quantitatively evaluated
    - Needed to ensure that sufficient margin exists to 10 CFR 100.11 dose reference values



Maximum Hypothetical Accident: Narrative (1 of 2)

The shutdown and heat removal systems are assumed to perform their safety functions but are not modeled. Instead, **hypothetical temperature curves** are used to conservatively drive radionuclide movement through the **functional containment**. Individual release pathways are discussed on the next slide.



Maximum Hypothetical Accident: Narrative (2 of 2)

- Radionuclides are postulated to diffuse from TRISO particles
  - The distribution of TRISO particles account for both manufacturing defects and in-service failures
  - Pre-transient diffusion of radionuclides from the fuel kernels are hypothetically and conservatively not modeled to maximize fuel inventory for release
- Radionuclides are postulated to evaporate and degas from the Flibe driven by conservative natural convection boundary conditions. No holdup of gases in Flibe is credited.
- Tritium is conservatively assessed to maximize both its inventory and release
  - The initial inventory of tritium is conservatively assessed
  - The release of tritium is conservatively postulated to:
    - desorb from in-vessel graphite as a function of temperature
    - instantaneously release from both steel and Flibe
- The Ar-41 inventory that is held up by closed graphite pores is instantaneously released

MHA: Methodology (1 of 3)

The Hermes MHA uses the methodology from the approved KP-FHR Mechanistic Source Term Methodology Topical (KP-TR-012-P-A). The following concepts directly leverage the topical report:

- Radionuclide grouping and transport approaches for the TRISO Fuel and Flibe coolant
- Mass transfer correlations for tritium into graphite reflectors and pebbles
- The gas space is not credited for confinement of the radionuclides that release from the Flibe free-surface
- “Two-hour holdup” assumptions for radionuclides transporting through the reactor building
- Conservative, unfiltered, ground level releases are modeled to maximize offsite doses

MHA: Methodology (2 of 3)

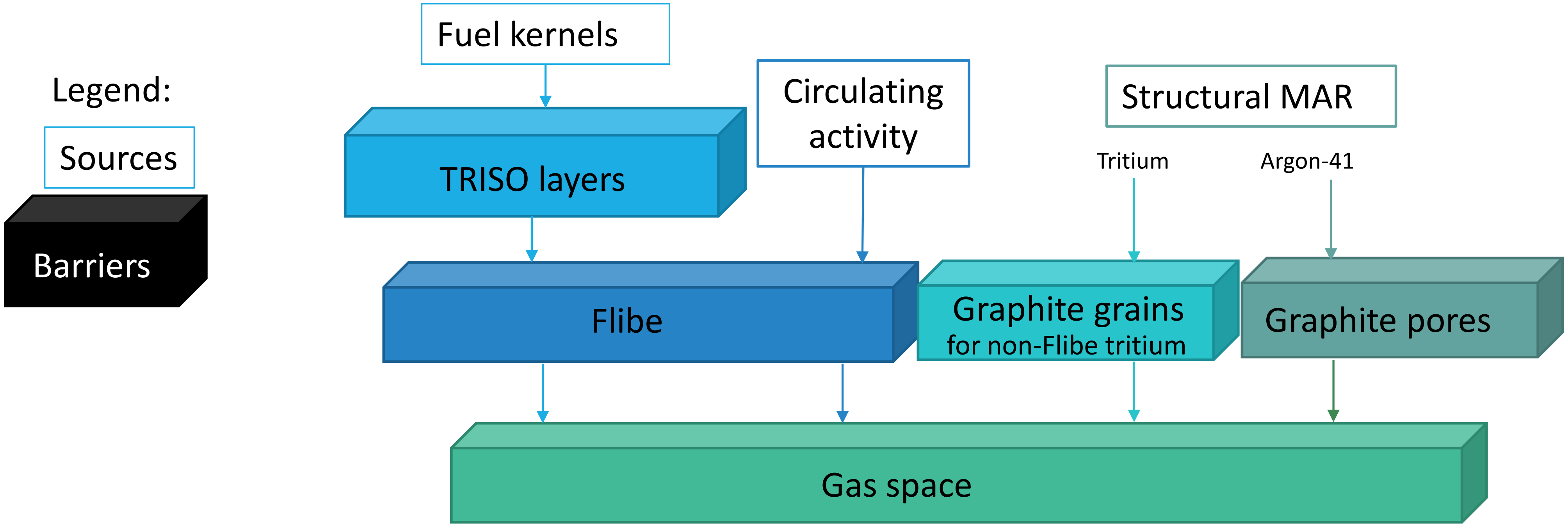
The following non-physical conditions provide additional hypothetical challenges to the functional containment (beyond what is described in KP-TR-012-P-A):

- Prescribed hypothetical temperature histories are applied to the transient. This ensures that the MHA will bound the system temperatures from the postulated event groups.
- Pre-transient diffusion of radionuclides from the fuel in the reactor core is neglected. This ensures that the maximum inventory is available for release at the initiation of the transient.
- A bounding vessel void fraction is assumed to facilitate the release of low volatility species in the vessel via bubble burst.
- Additional conservatism in tritium modeling to address limitations associated with tritium modeling in graphite is described in KP-TR-012-P-A.

MHA: Methodology (3 of 3)

1. Identify and account for the sources of material at risk (MAR) and the barriers to release
2. Evaluate release fractions for every combination of barrier, radionuclide group, and time interval
3. RADTRAD and ARCON evaluate dose consequences at the exclusion area boundary (EAB) and the low population zone (LPZ)

Three sources of MAR and associated release barriers



Maximum Hypothetical Accident:
Sources of MAR (1 of 2)

1. TRISO Fuel
    - Serpent 2 evaluation provides fuel inventory
    - Pre-transient depletion of radionuclides from the fuel is neglected to maximize inventory available for release

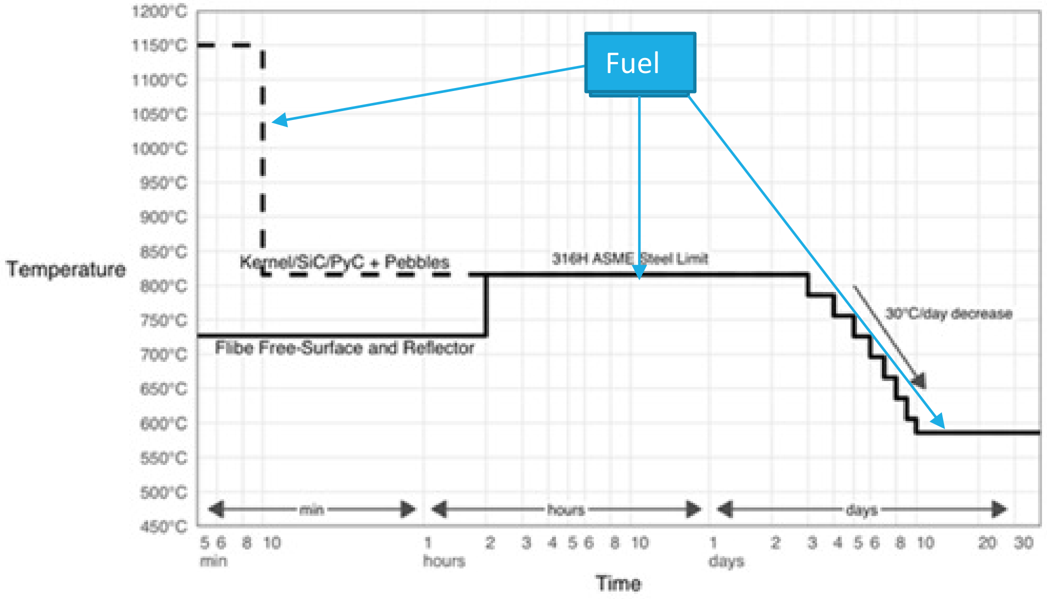
2. Circulating Activity
    - Bounding value of circulating activity is assumed in the analysis
    - Expected to be a variable controlled by technical specification

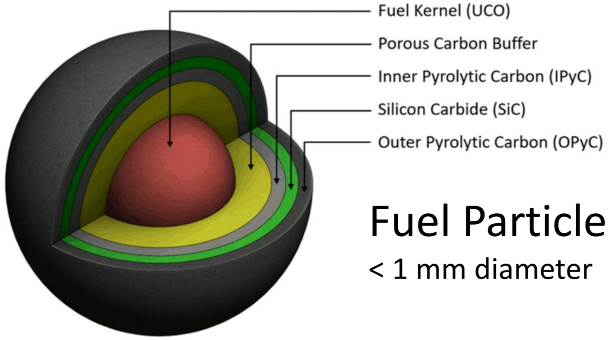
Maximum Hypothetical Accident: Sources of MAR (2 of 2)

3. Structural (steel, reflector, pebbles)
   - Tritium
     - The inventory conservatively bounds the operating lifetime at full capacity factor with margin while accounting for differential uptake rates for pebbles and reflector
     - Transfer from Flibe to structures
       - Born in the Flibe but transferred to and sorbed in structures (primarily graphite)
       - Transport speciation is conservatively assigned as tritium fluoride to maximize tritium sorption
       - Transfer from Flibe to structures determined by mass transfer coefficients from Flibe flow characteristics
     - Sorption within structures
       - Sorbed solely as a function of mass transfer from the Flibe to structures (i.e., no diffusion resistance)
       - Retained without modeling steady state release mechanisms (i.e., perfect absorber)
   - Argon-41
     - Produced via neutron activation of Ar-40 to Ar-41
     - The inventory available for release consists of Ar-41 contained within the graphite’s closed porosity

Maximum Hypothetical Accident: Release models for the TRISO Fuel

- Transport through TRISO fuel layers is modeled using Fick’s laws of diffusion
  - The CORSOR model is used for kernel diffusion
  - IAEA correlations are used for layer diffusion
- Diffusion is driven by the fuel’s hypothetical temperature curve
- Transient diffusion of fission products:
  - Is negligible if even a single PyC layer remains intact
  - Total releases are thus dominated by releases from exposed kernels



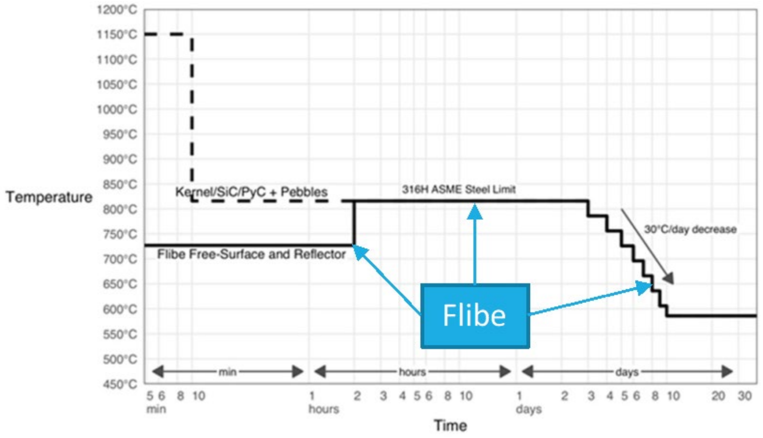


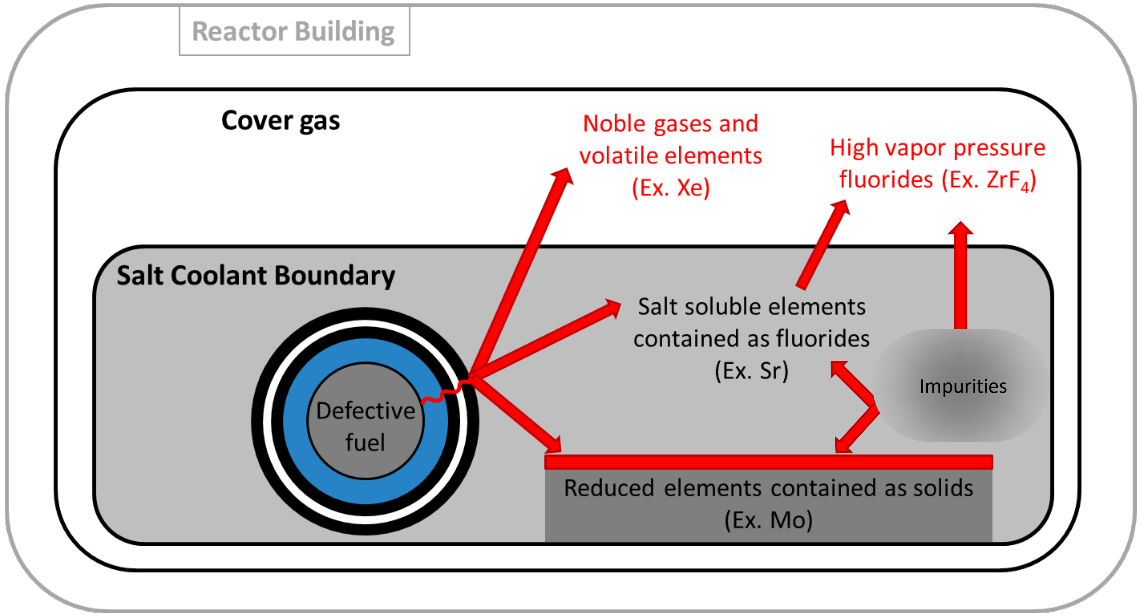
Fuel Particle

< 1 mm diameter

Maximum Hypothetical Accident: Release models for the Flibe Coolant

- Flibe provides a secondary functional containment barrier to:
  - Bounding circulating activity
  - In-transient release of fission products from TRISO
- Two release mechanisms are modeled for Flibe
  - Bubble burst
  - Evaporation




Reactor Building

Cover gas

Noble gases and volatile elements (Ex. Xe)

High vapor pressure fluorides (Ex. ZrF_4)

Salt Coolant Boundary

Salt soluble elements contained as fluorides (Ex. Sr)

Impurities

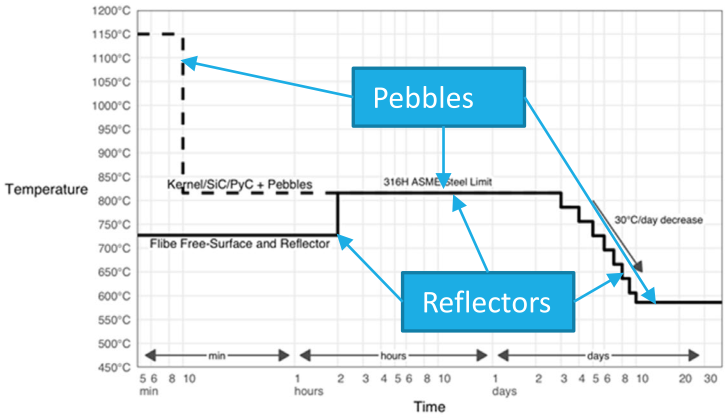
Defective fuel

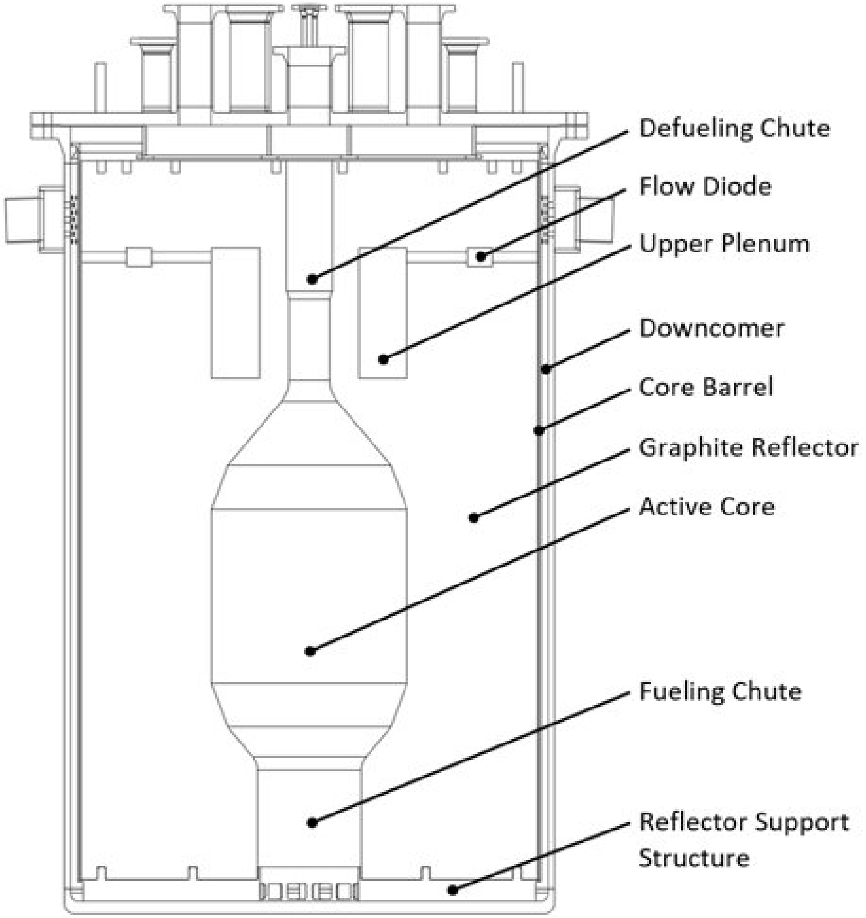
Reduced elements contained as solids (Ex. Mo)

- Certain radionuclide groups bypass the Flibe's functional containment
  - No credit for gas retention
  - High volatility noble metals evaporate near instantaneously

Maximum Hypothetical Accident: Release models for the Structural MAR

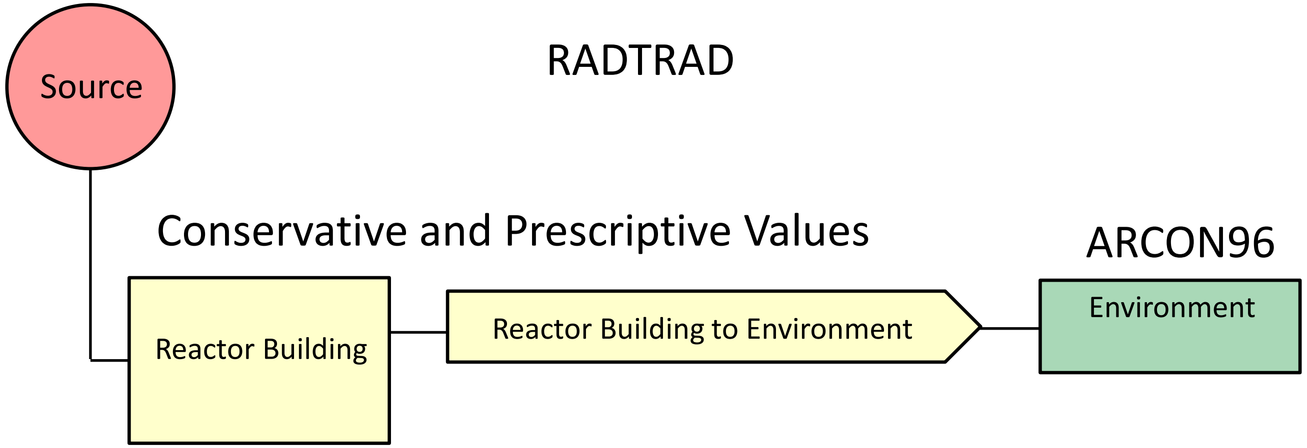
- Tritium in graphite grains
  - No-holdup of tritium in the Flibe instantly drops the concentration of tritium outside of graphite grains drops to zero
  - Tritium rapidly diffuses out of the graphite grain due to the non-physical concentration gradient
- MAR outside of graphite grains (e.g., steel, pores) are instantly released at the start of the transient





Maximum Hypothetical Accident: Release models for the Gas/Atmospheric Transport

- RADTRAD:
  - Input: Mobilized material-at-risk activities
  - Depletion mechanisms
    - Radioactive decay
    - Aerosol settling (i.e., Henry correlation)
  - Leakage rates (two-hour holdup)
- ARCON96:
  - Inputs
    - Hourly Meteorological Data
    - Distance from the reactor building to the following areas:
      - Exclusion area boundary
      - Low population zone
    - Approved values from KP-TR-012
  - Outputs
    - Time averaged dispersion values


RADTRAD

Conservative and Prescriptive Values

ARCON96

ARCON96

Distance (m)	$\frac{\chi}{Q}$ (s/m^3)				
	0-2 hrs	2-8 hrs	8 hrs – 1 day	1 – 4 days	4 – 30 days
250	1.51×10^{-4}	N/A	N/A	N/A	N/A
800	3.61×10^{-5}	3.51×10^{-5}	1.45×10^{-5}	1.54×10^{-5}	1.49×10^{-5}

Maximum Hypothetical Accident: Dose Consequences

Dose results meet 10 CFR 100.11 reference values at the EAB and LPZ with significant margin

Location and Duration	**Whole Body Dose (rem)**		**Thyroid Dose (rem)**	
	10 CFR 100	MHA Result	10 CFR 100	MHA Result
Exclusion Area Boundary (First 2 hrs at 250m)	25	0.227	300	0.235
Low Population Zone (30 days at 800m)	25	0.059	300	0.081

Summary

- The MHA dose consequence results meet the site dose reference values in 10 CFR 100.11(a)(1-2) at the EAB and LPZ with significant margin
- The MHA dose is bounding because it employs non-physical conditions that are beyond the design basis

Enclosure 2
Presentation Slides for the April 19, 2023 ACRS Kairos Power Subcommittee Meeting
(Non-Proprietary)

Hermes PSAR Chapter 13 Accident Analysis

DR. MATTHEW DENMAN – DISTINGUISHED ENGINEER, RELIABILITY

DR. TIMOTHY DRZEWIECKI – MANAGER, SAFETY ANALYSIS

ACRS KAIROS POWER SUBCOMMITTEE MEETING

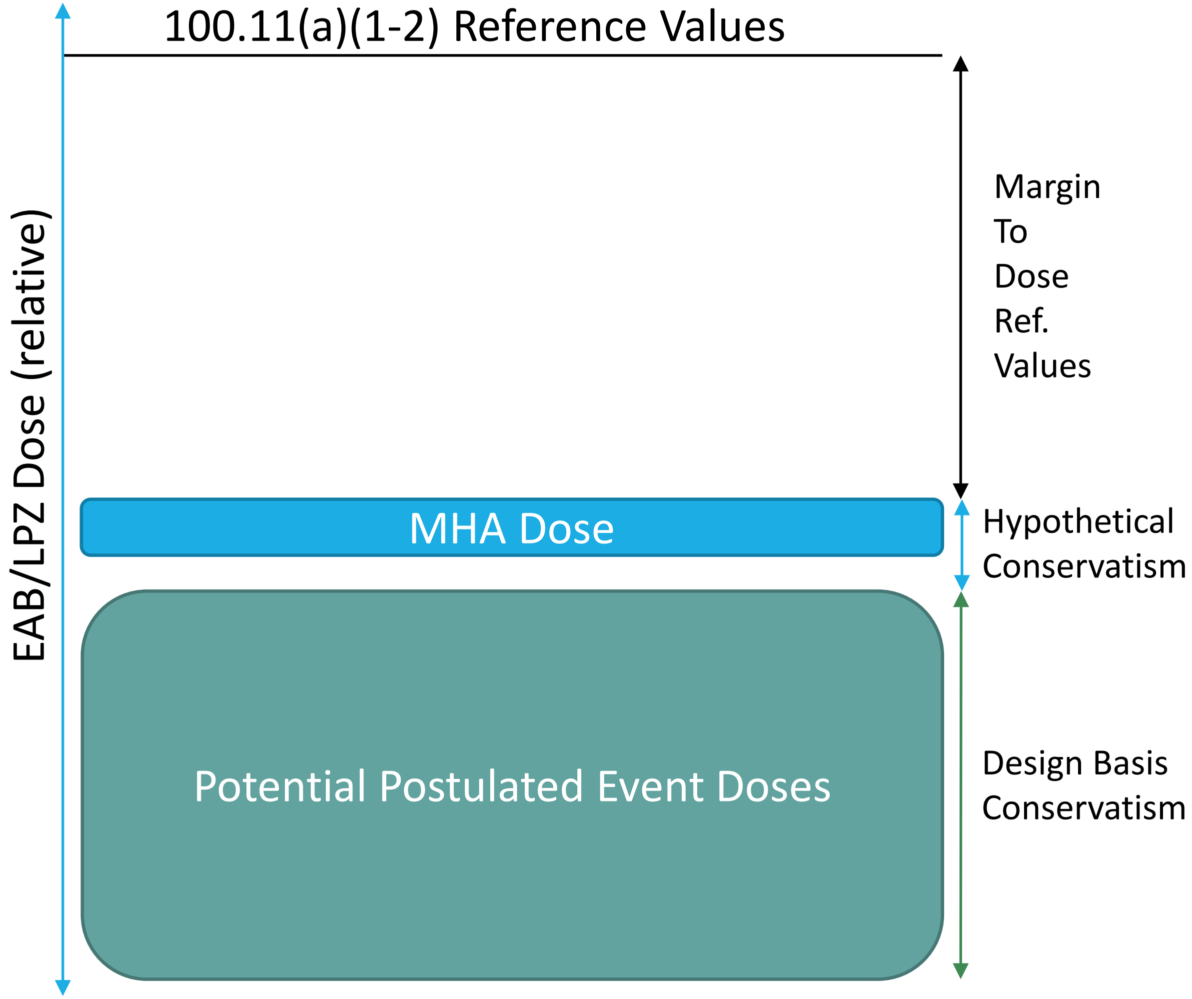
APRIL 19, 2023

Safety Case Summary

- 10 CFR 50.34(a)(4) requires a preliminary safety analysis to assess the risk to public health and safety from operation of the facility, including determination of the margins of safety
- To demonstrate compliance with 10 CFR 100.11 dose reference values, a Maximum Hypothetical Accident (MHA) that bounds the postulated events is analyzed for dose consequences by challenging the performance of functional containment
  - The Hermes MHA approach is consistent with guidance in NUREG-1537
  - The Hermes MHA is not physical
  - The Hermes MHA includes conservatisms that maximize source term
  - The Hermes MHA includes a postulated release of radionuclides
- To ensure that the postulated events are bounded by the MHA:
  - The list of postulated events is comprehensive to ensure that any event with the potential for significant radiological consequences has been considered
    - Initiating events and scenarios are grouped, so that a limiting case for each group can be qualitatively described in CPA (quantitative results will be provided with OLA)
    - Acceptance criteria are provided for the important figures of merit in each postulated event group to ensure the potential consequences of that event group remain bounded by the MHA as the design progresses
  - Prevention of an event initiator is justified in the PSAR

Relationship between Dose Limits, the Maximum Hypothetical Accident, and Postulated Events

- The Maximum Hypothetical Accident (MHA) is constructed to:
    - Be conservatively non-physical to overestimate potential off-site dose consequences
    - Provide confidence that sufficient safety margin exists
    - Ensure that reasonable design constraints will result in bounded postulated event doses
- In PSAR Chapter 13, the MHA dose is:
    - Quantitatively evaluated
    - Ensures that sufficient margin exists to 10 CFR 100.11 dose reference values



Postulated Event Analysis Methodology

- Postulated events are identified in Chapter 13 of the PSAR
  - Postulated events include any potential upset to plant operations, within the plant design basis, that causes an unplanned transient to occur
  - Justification is provided for those events excluded from the design basis (Prevented Events, PSAR Section 13.1.10)
- Figures of merit provide the means to measure and demonstrate that the resulting dose of a postulated event is bounded by the dose consequences of the MHA
- The preliminary methods and sample calculations of the postulated event groups are provided in KP-TR-018, Rev. 2. The methodology describes:
  - How to analyze figures of merit for each postulated event group
  - How the acceptance criteria ensure that the off-site dose consequences of postulated events are bounded by the MHA
- The final safety analysis results will be provided with the Operating License Application (including verification and validation of the evaluation models used)

Postulated Event Analysis Methodology (cont.)

- The evaluation model development activities for the postulated events follow a process similar to the Evaluation Model Development and Assessment Process (EMDAP) from Reg. Guide 1.203
- Postulated events with similar characteristics and modeling approaches are grouped into categories, consistent with NUREG-1537
- The limiting event for each event category is identified and qualitatively assessed from event initiation until a safe state is reached
- The safe state is defined in the methods for each category of events as a point where the transient figures of merit have stabilized in a safe condition

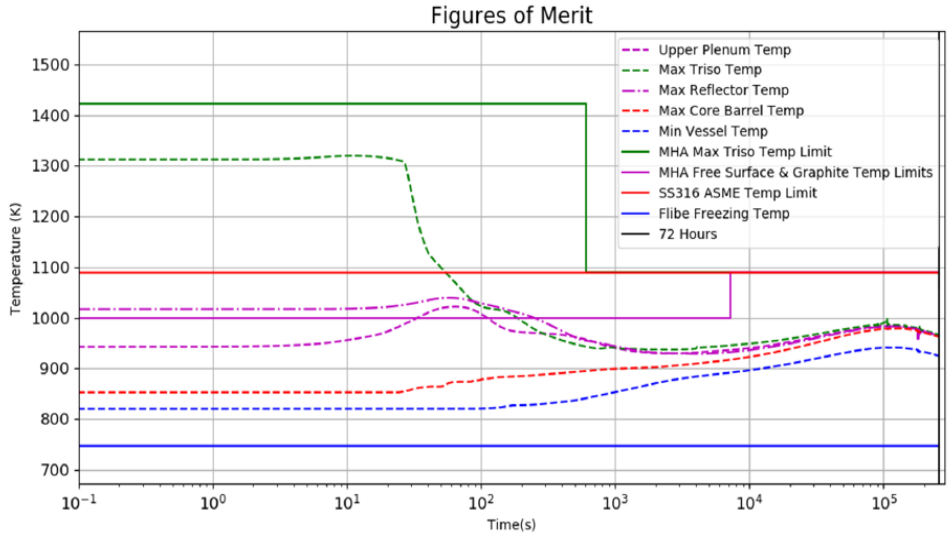
Input Parameters for Postulated Event Analysis

- Input parameters considered for postulated event analysis include a range of values to be evaluated for the final design (Table 4-4 of KP-TR-018)
- A range of values are assessed to identify the limiting scenario for each postulated event
- Key model uncertainties and initial conditions are conservatively applied to the methods to ensure figures of merit are conservatively predicted

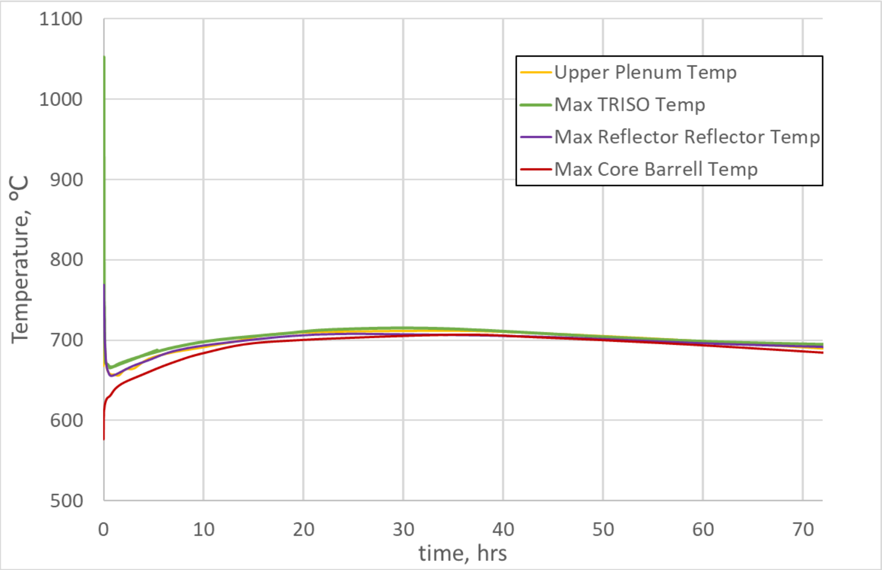
Limiting Postulated Events (1 of 3)

- Loss of Forced Circulation
  - Pump seizure disables the primary salt pump
  - Reactor protection system detects high coolant temperature and initiates a reactor trip
  - Grouped events include locked rotor and loss of normal heat sink
- Insertion of Excess Reactivity
  - Control system or operator error causes highest worth control element to withdraw continuously at the maximum control element drive speed
  - Reactor protection system detects the reactivity insertion due to a high neutron flux or high coolant temperature and initiates a reactor trip
  - Grouped events include fuel loading error, reflector shifting, and venting of gas bubbles
- General Challenges to Normal Operation
  - Includes challenges to normal operation not covered by another event category that require automatic or manual shutdown of the reactor
  - Bounded by the limiting loss of forced circulation postulated event
  - Grouped events include spurious trips, operator errors, and equipment failures

Sample Transient Analysis – Loss of Forced Circulation (Overheating)



KP-TR-018, Figure A4-1



Adapted from KP-TR-018, Figure A4-1

- Loss of forced circulation initiated by pump seizure/locked rotor
- Reactor trip on high plenum temperature reached ~30 seconds into event
- A second peak occurs ~20 hours into event followed by monotonic temperature decrease

Limiting Postulated Events (2 of 3)

- Mishandling or Malfunction of Pebble Handling and Storage System
  - Break in a fuel transfer line during removal from the core results in a spill of pebbles within the transfer line to the room
  - The reactor protection system detects this condition and initiates a pebble handling and storage system trip
  - Grouped events include transfer line break when pebbles are inserted into empty core, core at power, storage canisters, and mishandling of fuel outside the reactor
- Radioactive Release from a Subsystem or Component
  - Limiting event assumed to be a seismic event that results in a failure of all systems containing radioactive material that are not qualified to maintain structural integrity during a design basis earthquake
  - Design requirement on the amount of MAR for SSCs to be below the amount of MAR derived from the MHA
  - Grouped events include releases from the tritium management system, inert gas system, chemistry control system, and inventory management system

Limiting Postulated Events (3 of 3)

- Salt Spills
  - A hypothetical double-ended guillotine break occurs in the PHTS hot leg piping
  - Reactor protection system detects the salt spill due to a low coolant level and initiates a reactor trip
  - Grouped events include spurious draining of the PHTS, leaks from other Flibe containing systems, mechanical impact or collision of Flibe bearing SSCs, and HRR tube breaks
- Internal and External Hazard Events
  - Internal and external events include internal fire, internal water flood, seismic event, high wind, toxic release, mechanical impact or collision with SSCs, and external flood as described in Chapter 2
  - Events in this category are bounded by or considered as initiators in other event categories

Summary

Postulated events within the design basis are identified and grouped by characteristics and modeling approaches

- Design features which are credited with mitigating the effects of postulated events are described
- Figures of merit are derived for the postulated events to provide surrogate metrics which demonstrate that the resulting doses are bounded by the dose consequences of the MHA analysis
- The acceptance criteria for these figures of merit represent design limits that ensure the MHA is bounding

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

NRC Staff Review for PSAR Section 12.9 and Chapter 13

Briefing for the Advisory Committee on Reactor Safeguards

April 18-19, 2023

Office of Nuclear Reactor Regulation

Agenda

- PSAR Section 12.9, “Quality Assurance”
- PSAR Chapter 13, “Accident Analyses”
  - Maximum Hypothetical Accident (MHA) – PSAR Sections 13.1.1 and 13.2.1
  - Postulated Events and Other Sections – PSAR Sections 13.1.2 to 13.1.10 and 13.2.2
- Common Agenda for Each Chapter
  - Overview of PSAR Chapter and Principal Design Criteria (PDC)
  - Referenced topical reports (if applicable)
  - Staff technical evaluation
  - Findings and conclusions

Common 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.”
- **Guidance:** NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 2, “Standard Review Plan and Acceptance Criteria.”

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Protecting People and the Environment

NRC Staff Review for PSAR Section 12.9 Quality Assurance

Briefing for the Advisory Committee on Reactor Safeguards

April 18, 2023

By the Division of Reactor Oversight
Office of Nuclear Reactor Regulation

Overview of PSAR Section 12.9, "Quality Assurance"

- PSAR Section 12.9, "Quality Assurance" states that the description of Kairos' quality assurance (QA) program for the design, construction, and operation of Hermes is based on:
  - Regulatory Guide (RG) 2.5, "Quality Assurance Program Requirements for Research and Test Reactors," Revision 1, which endorses:
    - American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.8-1995, "Quality Assurance Program Requirements for Research Reactors."
- Kairos provided its Quality Assurance Program Description (QAPD) as Appendix B to PSAR Chapter 12 (i.e., PSAR Appendix 12B).

Additional Regulatory Guidance

- NRC RG 2.5, “Quality Assurance Program Requirements for Research and Test Reactors,” Revision 1 endorses:
  - ANSI/ANS 15.8-1995, “Quality Assurance Program Requirements for Research Reactors.”

Staff Evaluation

- Reviewed QAPD using the guidance in ANSI/ANS 15.8-1995, which is endorsed by NRC RG 2.5, Revision 1.
  - Evaluated QAPD Section 1, “Introduction,” and Section 2, “Design, Construction, and Modifications,” for the issuance of a construction permit (CP) because those sections apply to Hermes’ design, fabrication, construction, and testing.
    - Staff found Kairos followed ANSI/ANS 15.8-1995 in most sections. The following slides focus on areas where Kairos deviated from the standard.
  - Did not evaluate QAPD Section 3, “Facility Operations,” because it would apply to Hermes’ operation and is therefore not relevant to the issuance of a CP.

Staff Evaluation

- Staff evaluation of QAPD sections not in accordance with ANSI/ANS 15.8-1995:
  - Kairos proposed an alternate definition of "safety-related" to match the definition of "safety-related" in PSAR Chapter 3
    - The staff found this to be acceptable
  - PSAR Appendix 12B, Section 2.19 – Experimental Equipment:
    - Kairos did not provide description of controls for experimental equipment
    - PSAR Section 10.1 states that Hermes will not include special facilities dedicated to the conduct of reactor experiments or experimental programs.
    - Based on this, the NRC staff finds it acceptable that the QAPD does not include controls for experimental equipment.

Staff Evaluation (continued)

- Staff evaluation of QAPD sections not in accordance with ANSI/ANS 15.8-1995:
  - PSAR Appendix 12B, Section 4 – Applicability to Existing Facilities & Section 5 – Decommissioning:
    - ANSI/ANS-15.8-1995, Sections 4 and 5, are not applicable to the Hermes CP application
    - Acceptable that the QAPD did not include this recommended information, because Kairos did not indicate that the QAPD will apply to any existing facilities, and because submission of decommissioning plans and associated quality assurance provisions is not required until a licensee applies for license termination after permanent cessation of operations.

Recommended Construction Permit Condition

- The staff recommends that the construction permit include the following condition:
  - Kairos shall implement the QA program described, pursuant to 10 CFR 50.34(a)(7), in Chapter 12, Appendix B, of Revision 2 of the Hermes PSAR, including revisions to the QA program in accordance with the provisions below:
    - Kairos may make changes to its previously accepted QA program description without prior NRC approval, provided the changes do not reduce the commitments in the QA program description as accepted by the NRC.
      - Changes to the QA program description that do not reduce the commitments must be submitted to the NRC within 90 days.
    - Changes to the QA program description that do reduce the commitments must be submitted to the NRC and receive NRC approval prior to implementation.

Conclusion

- NRC staff finds the preliminary design information is consistent with the applicable criteria in NUREG-1537

- The staff concludes information in Hermes PSAR Section 12.9 and Appendix 12B is sufficient for the issuance of a CP in accordance with 10 CFR 50.35 and 50.40
  - Further information as may be required to complete the review of Kairos’s QA program for the conduct of operations and decommissioning can be reasonably left for the OL application.

Questions?

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NRC Staff Review for PSAR Section 13 Accident Analysis – Maximum Hypothetical Accident

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Briefing for the Advisory Committee on Reactor Safeguards

April 18, 2023

By the Division of Advanced Reactors and
Non-Power Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Overview of PSAR 13 – Maximum Hypothetical Accident (MHA)

- Preliminary analysis
- Consequences bounding for postulated events
- MHA event description and assumptions in PSAR Section 13.1.1
- MHA consequence analysis in PSAR Section 13.2.1
- PSAR Section 13.2.2 describes the postulated event methodology and assurance that the MHA consequence analysis is bounding for postulated events
  - Staff's evaluation will be presented tomorrow

Referenced Topical Reports

- KP-TR-011-NP-A, Revision 1, “Fuel Qualification Methodology for the Kairos Power Fluoride-Salt Cooled High Temperature Reactor (KP-FHR)”

- KP-TR-012-NP-A, Revision 1, “Mechanistic Source Term Methodology for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor”

MHA Description and Assumptions

- PSAR Figure 13.2-1 MHA hypothetical temperature vs. time profile to give bounding radionuclide releases
  - Not based on a specific scenario
  - Fission product release and transport mainly through diffusion driven by temperature
  - Final determination that temperature vs. time curve is conservative to postulated events at the OL review
- Assumes safety-related systems function as designed but includes consideration of the single failure criterion
- No incremental fuel particle coating failures from transient

MHA Consequence Analysis

- Accident source term methodology
  - Model system as sources of radioactive material at risk of release (MAR) and barriers to release
  - Apply release fraction to each barrier to eventually result in release to environment
    - Consistent with functional containment
  - Gravitational settling of Flibe aerosols in reactor building

MHA Source Term Modeling

- Radionuclide diffusion releases from fuel, Flibe, and graphite as a function of hypothetical temperature vs. time profile in PSAR Figure 13.2-1
- MHA assumes conservative fuel, Flibe, structural, and cover gas releases
  - The complete fuel inventory is available for release into the Flibe
  - Bounding failed fuel fractions by cohort are assumed
  - Flibe and cover gas radionuclide inventories are set to technical specification values
- Except for fuel transient releases, tritium, and Argon-41, the MHA uses approved mechanistic source term (MST) models from KP-TR-012-P-A
  - Fuel releases are modelled using accepted methods
  - Staff reviewed the fuel release references and found the models acceptable
  - Tritium modeling resulting in higher total releases
  - Staff evaluated the modeling assumptions for Ar-41 and found them to be conservative
    - Audit of Ar-41 calculation

Staff Evaluation – MHA Consequence Analysis

- Preliminary MHA dose analysis methods and assumptions are consistent with the approved MST methodology KP-TR-012-P-A
  - Staff will review details of final implementation in OL

- Staff evaluation of the site-characteristic short-term atmospheric dispersion factors is discussed in SE Section 2.3

Staff Evaluation – MHA Consequence Analysis

- Staff audit of preliminary consequence analysis and MHA source term information
  - Confirmed PSAR description of MHA analysis
  - Kairos calculations and reference reports
    - Initial radionuclide inventory/MAR sources
      - Fuel, Flibe
      - Tritium and Ar-41 inventories in graphite
    - Releases from graphite
    - Modeling of radionuclide transport across barriers/release fractions
  - In-person discussion with Kairos staff showing example of how to take information from the MHA to develop the RADTRAD code input to calculate doses

Staff Evaluation Findings – MHA

- The MHA serves as a bounding hypothetical analysis for Hermes
- The combination of bounding conditions analyzed are beyond what is assumed for postulated events
- Preliminary dose results for MHA are substantially below the regulatory dose reference values for test reactor siting in 10 CFR 100.11
- Because assumptions of the MHA are bounding, calculated doses will likely not be exceeded by any accident considered credible
  - Staff will confirm calculations as part of the OL application review

Staff Evaluation – Control Room Habitability

- SE Section 13.2.1 also includes staff evaluation of preliminary information on control room radiological habitability described in PSAR Section 7.4
  - Identifies relevant design basis as PDC 19
  - Additional description of the control room habitability design and dose analysis corresponding to the final detailed design will be provided in the OL application

Conclusion

- NRC staff finds the preliminary design information and analysis are consistent with the applicable criteria in NUREG-1537

- The staff concludes information in Hermes PSAR Chapter 13 on the MHA is sufficient for the issuance of a CP 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 Chapter 13 Accident Analysis – Postulated Events and Other Sections

Briefing for the Advisory Committee on Reactor Safeguards

April 19, 2023

Office of Nuclear Reactor Regulation and
Office of Nuclear Regulatory Research

Overview of PSAR Chapter 13, “Accident Analysis”

- Kairos uses a Maximum Hypothetical Accident (MHA) approach to bound other postulated events
  - Postulated events are bounded by the MHA radionuclide release
- Approach uses figure of merit and acceptance criteria in PSAR Table 13.1-1 to ensure MHA remain bounding if different radionuclide release pathways exist
- Postulated events considered are consistent with those listed in NUREG-1537
  - Some technology-specific events or event sequences are precluded by design, such as Flibe interaction with concrete or water (e.g., decay heat removal system (DHRS) water leak)
  - Some technology-specific events (e.g., increased pebble bed packing fraction) have been evaluated

Postulated Event Methodology – Generic Aspects

- Postulated event methodology is described in technical report KP-TR-018, Rev 2
- KP-TR-018, Table 4-4, “Input Parameters for Postulated Events,” identifies parameters and their ranges to be considered for all Chapter 13 events
  - Examples: initial power level, reactor coolant temperature
  - The NRC staff finds that KP-TR-018 specifies acceptable ranges for parameters to ensure the most limiting scenarios are analyzed
- FSAR analyses will consider a full range of sensitivities based on Table 4-4
- KP-SAM and KP-BISON have the capability to model postulated events and the corresponding fuel releases
  - Code verification and validation will be reviewed prior to, or as part of, the OL application

PSAR Section 13.1.2, Insertion of Excess Reactivity Event

- PSAR analysis assumes continuous withdrawal of highest worth control element at the maximum speed
  - Reactor trips on high power or high outlet temperature
  - A range of reactivity insertion rates and initial core powers will be evaluated in the OL application
  - Uncertainties will be quantified as part of the OL application
  - Control element ejection is precluded due to the low differential pressure

- Temperatures stay below the assumed MHA hypothetical temperature vs. time curve except for the maximum reflector temperature, which slightly exceeds the MHA free surface and graphite temperature limits for a short period of time

- Staff's scoping analysis yielded similar results as will be shown in following slides

- The staff has reasonable assurance the MHA dose bounds that of the insertion of excess reactivity event because of conservatisms in the MHA analysis

PSAR Section 13.1.3, Salt Spill Event

- Salt spill is a loss of coolant inventory resulting in different release pathways than the MHA
  - Assumes safety-related systems work as intended
  - Assumes water or concentrate interactions are precluded by design
  - Methodology includes evaluating a range of break sizes and locations
- Release pathways different from the MHA include radionuclides mobilized by the break, evaporation from the spilled pool, and oxidation of any exposed graphite
- Heat up due to the loss inventory is expected to be low and bounded by the assumed MHA temperature vs. time curve

PSAR Section 13.1.3, Salt Spill Event (continued)

- Methodologies for break aerosol generation and Flibe vessel free-surface evaporation methodologies are from the approved mechanistic source term (MST) topical report
- Salt spill uses lower, event-specific temperatures, hence lower fuel, wetted graphite surface tritium, and lower Flibe vessel free-surface releases
- Staff has reasonable assurance that MHA would bound the salt spill event based on the minimum heat up and low salt mass spilled
- A quantitative dose comparison between the salt spill event and MHA will be performed in the OL application

PSAR Section 13.1.4, Loss of Forced Circulation

- PSAR analysis assumes a primary salt pump seizure
  - Reactor trips on high outlet temperature
  - Uncertainties will be quantified as part of the OL application

- Temperatures stay below the assumed MHA hypothetical temperature vs. time curve except for the maximum reflector temperature and upper plenum temperature, which slightly exceed the MHA free surface and graphite temperature limits for a short period of time

- Staff's scoping analysis yielded similar results, as will be shown in following slides

- The staff has reasonable assurance the MHA dose bounds that of the loss of forced circulation

PSAR Section 13.1.5, Pebble Handling and Storage System (PHSS) Event

- PHSS event is a break in a pebble handing line resulting in different release pathways than the MHA
  - Reactor protection system trips the PHSS to stop pebble movement
  - Pebbles spill into the transfer line room and no active heat removal (i.e., room HVAC) is credited to limit spilled pebbles temperature
  - Criticality is precluded by design and pebbles are assumed to remain intact

- Release pathways different from the MHA include radionuclides mobilized graphite dust from the break and pebble oxidation

- Spilled pebbles are assumed at their maximum burnup and hence pebble matrix material at risk is conservative for oxidation and dust activity determinations

PSAR Section 13.1.5, Pebble Handling and Storage System (PHSS) Event (continued)

- Staff reviewed methodologies for pebble matrix oxidation and dust generation rate and transport and found them acceptable
  - Fuel Qualification (FQ) topical report (KP-TR-011) states pebble matrix oxidation tests will be performed to validate the PSAR assumed oxidation correlation
  - FQ topical report states pebble wear against SS-316 will be tested to inform the PHSS dust generation rate
  - MST topical report Section 7.3.3.2.2. evaluates the dust resuspension methodology

- PHSS event uses lower temperatures (event specific) temperatures hence lower fuel, wetted graphite surface tritium and lower Flibe vessel free-surface releases

- A quantitative dose comparison is between the PHSS event and MHA will be performed in the OL application

PSAR Section 13.1.6, Radioactive Release from a Subsystem or Component

- Radioactive material at risk of release (MAR) is limited such that the release, assuming no retention, is bounded by the MHA
  - This includes all locations not qualified to maintain structural integrity during a postulated event (e.g., seismic event).
- Potential area with MAR limits include the tritium management system, inert gas system, chemistry control system (including filters), and inventory management system

PSAR Section 13.1.8, General Challenges to Normal Operation

- PSAR Section 13.1.8 addresses events caused by inadvertent operator action, failure of a control system or instrumentation

- The reactor protection system (RPS) will sense and terminate the event assuming the setpoints are reached

- Events caused by operator action, control system or instrument failures are typically bounded by the events analyzed in Chapter 13 due to the use of bounding assumptions and analysis

- Consequences caused by inadvertent operator action, control system or instrumentation failure will be reviewed in more detail as part of the OL application

PSAR Section 13.1.9, Internal or External Hazard Events

- Limiting internal events are primarily addressed by Chapter 13
  - Fire protection systems and programs are addressed in PSAR Section 9.4 and will protect safety-related SSCs that perform event mitigation
- Most external events are addressed by designing SSCs commensurate with the hazard or applicable standard
- A seismic-induced reactivity event is unique to pebble bed reactors
  - Potential increase in pebble packing fraction and associated reactivity increase expected to be bounded by the Chapter 13 insertion of excess reactivity event
  - Reactivity insertion due to pebble bed slumping (i.e., elevation change of the active core) is not expected in a buoyant molten salt pebble bed
  - Staff to review detailed seismic induced packing fraction reactivity analysis as part of the OL application review

PSAR Section 13.1.10, Prevented Events

- Prevented events are potential events which are prevented due to design features and hence are not evaluated

- Of the PSAR prevented events listed, the staff issued requests for additional information (RAI) on two of the prevented events
  - RAI 348 (ML22227A180) asked the basis for why recriticality and unprotected events are excluded from consideration
    - Kairos modified PSAR Section 4.2.2.3 to further describe the shutdown element insertion testing to ensure the shutdown margin analysis remains valid and, in part, to lower the probably of an unprotected event

  - RAI 350 (ML22227A192) asked, in part, how component integrity is ensured for the duration of an air ingress event including air ingress beyond the heat rejection blower trip
    - SE Section 5.1.3.2.6 addresses material qualification testing out to 7 days
      - Beyond 7 days compensatory measures could reasonably to be established to ensure the final reactor state protects public health and safety

Staff Scoping Analysis of Hermes

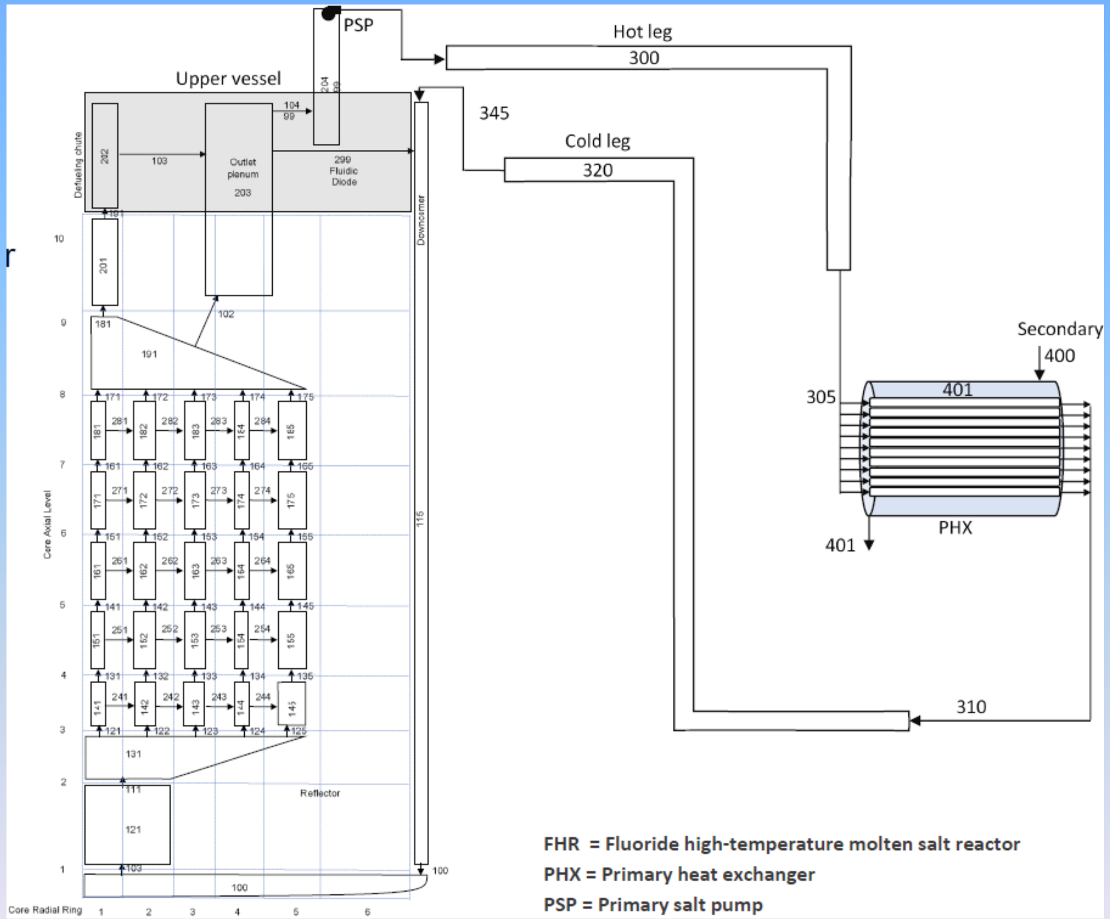
- NRC developed several ‘representative’ non-LWR systems models since 2020
  - Part of “Non-LWR Vision and Strategy, Volume 3” covering severe accidents/source term
  - Included UC Berkeley Mark 1 design, representing TRISO pebble fueled/molten salt cooled FHR
  - SCALE code suite used for inventory and reactor physics data generation (ORNL)
  - MELCOR used for accident progression using SCALE-produced data (Sandia)

MELCOR Analysis Approach

- Original SCALE/MELCOR FHR work used the UCB Mark I design and focused on fission product release from TRISO and molten salt during beyond design basis events
- The UCB Mark I model was modified (January-March 2022) for the Kairos Hermes design and applied to select Chapter 13 postulated events
  - Modifications based on PSAR information and engineering judgement
  - SCALE-generated inventory and decay heat input
  - Transients from technical report KP-TR-018: insertion of excess reactivity, loss of forced circulation

MELCOR Model Description

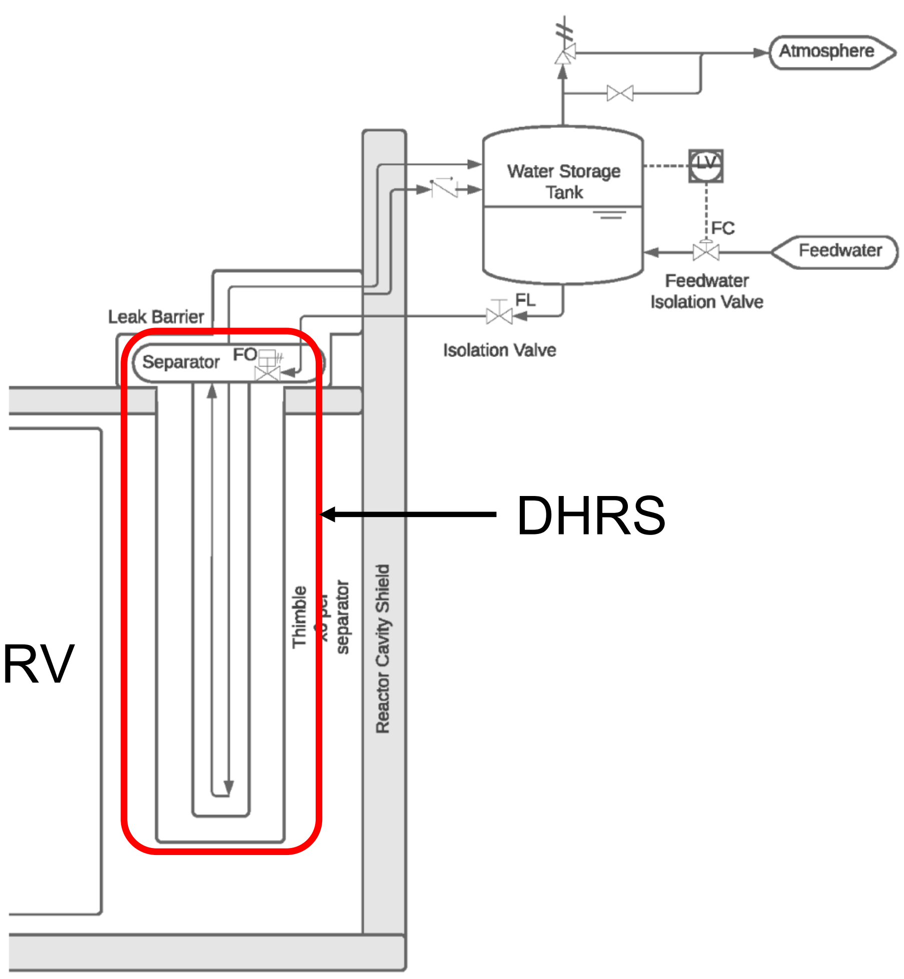
- Model focuses on primary system
  - Intermediate loop and DHRS represented via boundary conditions
  - Necessary given lack of detailed design info
- Detailed representation of flow paths within pebble bed
- Fluidic diodes represented as flow path with different forward and reverse loss coefficients



FHR = Fluoride high-temperature molten salt reactor
PHX = Primary heat exchanger
PSP = Primary salt pump

MELCOR DHRS Model Description

Figure 6.3-1: Functional Diagram of the DHRS

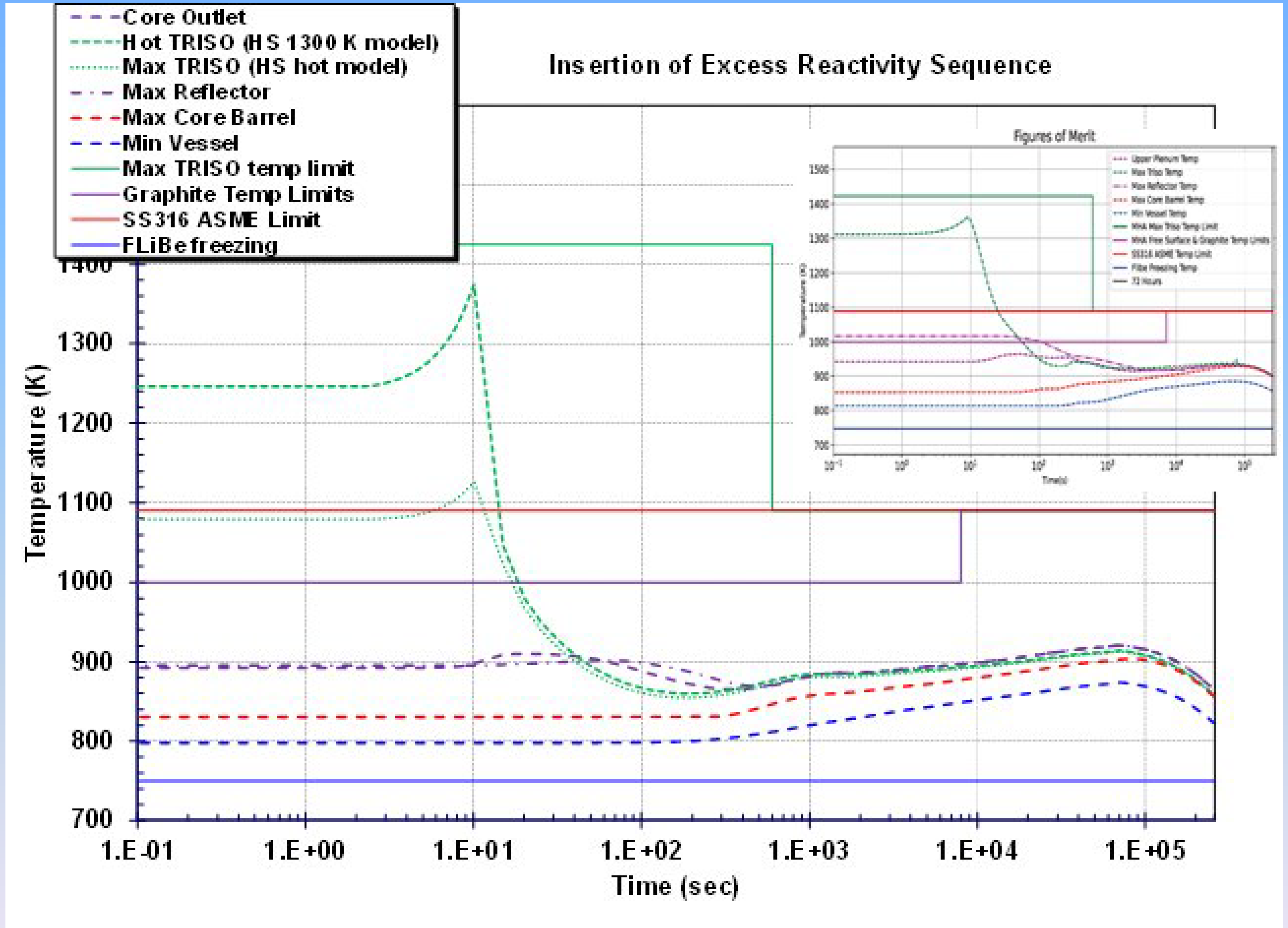


PSAR Schematic

- Heat transfer between the reactor vessel (RV), DHRS, and cavity wall
  - Multi-surface radiation enclosure model
  - Natural convection heat transfer from all surfaces
  - Surface emissivities (variable - uncertainty parameter)
  - Convective heat transfer coefficients (variable - uncertainty parameter)
- DHRS model
  - 0, 6, 12, 18, or 24 DHRS thimbles
  - Water (constant boundary condition at 100°C)
  - Water to DHRS evaporator tube wall uses boiling heat transfer coefficient
  - Thermal resistance between evaporator tube to thimble casing (variable - uncertainty parameter)
- Cavity wall
  - Fire brick, steel liner, concrete wall
  - No liner cooling

MELCOR Results: Insertion of Excess Reactivity

- Withdrawal of control element inserts 3.02$* over 100 seconds
- Reactor trips on high power (120%) at ~9 s, concurrent with PSP trip and flow coastdown
- Temperatures maintained within MHA envelope

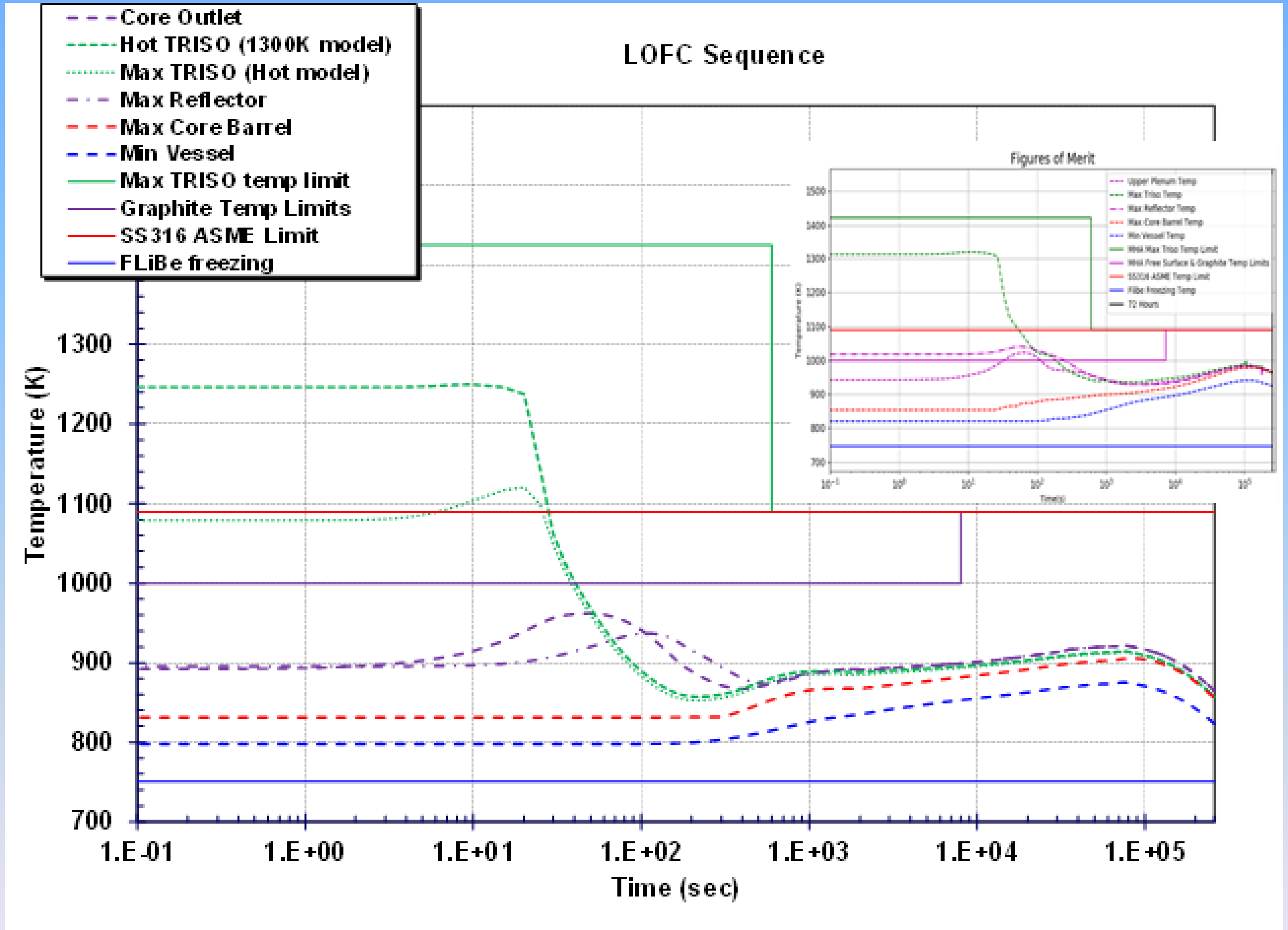


MELCOR results as compared with PSAR (upper right)

* "$" is a unit of reactivity used in nuclear reactor analysis.

MELCOR Results: Loss of Forced Circulation

- Concurrent trip of primary and intermediate coolant pumps results in flow coastdown
- Two cases presented:
  - Overheating
  - Overcooling (Flibe freezing)
- Reactor trips on overtemperature
- System remains within MHA envelope



MELCOR results for overheating case as compared with PSAR (upper right)

Overall Staff Conclusions

- The staff found the postulated event methodologies can be used to predict conservative event temperatures and dose releases
- The staff reviewed PSAR Table 13.1-1, “Acceptance Criteria for Figures of Merit” and found these acceptable as described SE Section 13.2.2 because they account for physical phenomena and additional release pathways that are not part of the MHA to ensure the MHA remains bounding
- OL application will provide dose analyses for events bounded by the MHA release, along with a comparison to the acceptance criteria for the figures of merit in PSAR Table 13.1-1

Overall Staff Conclusions

- Because the figures of merit and associated acceptance criteria ensure that the MHA releases remain bounding, the staff has reasonable assurance that the radiological consequences of the postulated events will also meet regulatory requirements of 10 CFR 100.11 and 10 CFR 50.34(a)

- The staff concludes information in the Hermes PSAR Chapter 13 is sufficient for the issuance of a CP in accordance with 10 CFR 50.35 and 50.40 and further information can be reasonably left for the OL application