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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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KAIROS POWER LICENSING SUBCOMMITTEE

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FRIDAY

NOVEMBER 19, 2021

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The Subcommittee met via Teleconference, at 9:00 a.m. EST, David A. Petti, Chair, presiding.

COMMITTEE MEMBERS:

DAVID A. PETTI, Chair

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member

JOSE MARCH-LEUBA, Member

JOY L. REMPE, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

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

9:00 a.m.

CHAIR PETTI: Well, good morning, everyone. The meeting will now come to order.

This is a meeting of the Kairos Power Licensing Subcommittee of the Advisory Committee on Reactor Safeguards. I am David Petti, Chairman of today's Subcommittee meeting.

ACRS members in attendance are Vicki Bier, Charles Brown, Jose March-Leuba, Joy Rempe, Ron Ballinger, Vesna Dimitrijevic. And I don't see anybody else.

Our consultants. Let's see. I don't see any of our consultants either at this point, but they may be in by phone.

Weidong Wang of the ACRS staff is the designated federal official for this meeting.

During today's meeting the Subcommittee will review staff Safety Evaluation Report on the KP-FHR Mechanistic Source Term Methodology, Revision 1. The Subcommittee will hear presentations by and hold discussions with the NRC staff, Kairos Power representatives, and other interested persons regarding this matter, but part of the presentations by the applicant and the NRC staff may be closed in order to discuss information that is proprietary to the licensee and its contractors pursuant to 5 U.S.C. 552(b)(C)(4).

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

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

The rules for participation in today's meetings have been announced as part of the notice of this meeting previously published in the Federal Register.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register notice.

Due to the COVID pandemic today's meeting is being held over Microsoft Teams for ACRS, NRC staff, and the licensee attendees. There is also a telephone bridge line allowing participation of the public over the phone.

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

We'll now proceed with the meeting. And I'd like to start by calling on William Kennedy, NRR management.

MR. KENNEDY: Well, good morning, Mr. Chairman and distinguished members of the Advisory Committee on Reactor Safeguards. My name is William Kennedy. I'm the Acting Chief of the Advanced Reactor Licensing Branch in NRR's Division of Advanced reactors and Non-Power Production and Utilization Facilities.

It's my pleasure to be here today to provide introductory remarks on behalf of the division. With me today are Ms. Michelle Hart, who is the lead technical reviewer. Mr. Alex Chereskin. They are both from the Advanced Reactor Technical Branch No. 2 in DANU. Mr. Jason White is here from the External Hazards Branch in the Division of Engineering and External Hazards. And all of them will be providing the staff presentation. We also have Mr. Samuel Cuadrado de Jesus who is providing project management support for the review of this topical report.

The staff is looking forward to discussions with and feedback from ACRS members today on the Draft Safety Evaluation of the Kairos Power topical report that's titled KP-FHR Mechanistic Source Term Methodology. So as you will hear this topical report is important for Kairos' development of accident source terms and atmospheric dispersion values for use in radiological consequence analysis for siting and safety analysis for Kairos Power's fluoride salt-cooled high-temperature reactor designs, also known as the KP-FHR designs.

The report also describes development of source terms for estimation of dose for anticipated operational occurrences in design-basis events to be used in the endorsed NEI 18-04 methodology for applicants to categorize events, classify and describe special treatment of structures, systems, and components, and assess defense-in-depth for non-light water reactors.

This topical report is related to other Kairos topical reports such as the Fuel Performance Methodology Report. Limitations and conditions on the use of the topical report are identified to ensure that the methods and underlying assumptions are applicable to the specific design in future KP-FHR license applications.

So I'd just like to note that this is the fourth time the staff and Kairos Power have had the opportunity to brief ACRS on Kairos' topical reports and so the staff appreciated the helpful comments from the ACRS on the recent topical report evaluation covering reactor coolant scaling methodologies, licensing modernization project implementation, and most recently the Fuel Performance Methodology Report.

Staff looks forward to continuing to work with Chairman Petti and the rest of the ACRS members and staff as we complete reviews of more Kairos Power topical reports and review license applications for facilities that will use the Kairos Power design.

In September we received a construction permit application for the Kairos Power Hermes test reactor, and that is currently being reviewed for acceptance.

I'd also like to highlight the working relationship between the NRC staff and Kairos Power has been excellent. Similar to previous reviews of Kairos Power topical reports the staff and Kairos have used public meetings as an efficient means for addressing technical issues without the need for significant formal requests for additional information.

And then finally I'd like to give a big thanks to the technical staff for their efforts to produce a high-quality Draft Safety Evaluation Report and also for project management of this review.

So that concludes my opening remarks. Thank you very much.

CHAIR PETTI: Thank you. Before we turn it over to Kairos I just want to note for the record that our consultant Steve Schultz has joined us.

So, Kairos, the floor is yours.

MEMBER BROWN: Dave, are they showing their slides?

CHAIR PETTI: Not yet.

Kairos, are you out there?

MEMBER BROWN: Just wanted to make sure I wasn't the only one.

MR. PEEBLES: Okay. Sorry. We were having some technical difficulties with the conference room.

Thank you, Mr. Chairman, and good morning, everyone. My name is Drew Peebles. I'm the Manager of Licensing and Safety Integration here at Kairos Power. Before we get started I would like to thank the ACRS members (audio interference).

CHAIR PETTI: Okay. I don't hear them anymore. Do other people have that problem?

MEMBER BROWN: It sounds like we've lost them, Dave.

MEMBER MARCH-LEUBA: Yes, like -- yes, they were having -- we were trying to test their conference room yesterday. They were having some technical issues.

CHAIR PETTI: Okay.

(Pause.)

CHAIR PETTI: Okay. I hear you guys

again.

MR. PEEBLES: Sorry about that. So I was saying we would like to thank the ACRS members for your continued interest in Kairos Power. William Kennedy mentioned that we've had four briefings in front of the ACRS to date. I believe this is the fifth topical that we will bring to you on.

(Audio interference.)

CHAIR PETTI: And they've faded out again.

So some of the -- I'm assuming some of the Kairos folks that I see listed that may not be in the conference room are texting them and telling them.

(Pause.)

MR. PEEBLES: Sorry about this. So I think I mentioned that William Kennedy also mentioned that we were -- that we've briefed the ACRS four times to date and I believe this is the fifth topical that we get a chance (audio interference).

CHAIR PETTI: Okay. We're continuing to have problems. I'm wondering if I should go out and come back in, if that would help.

MR. PEEBLES: Okay. Can you hear us now?

CHAIR PETTI: A little echo, but yes. Oh, a big echo.

MR. PEEBLES: Okay. We've joined with a

different laptop. Sorry about the conference room issues.

So as I was mentioning, we have recently submitted our construction permit application for our non-power reactor that we refer to as Hermes and we look forward to engaging with the ACRS in the review of that application as well.

I would also like to thank the NRC staff for a thorough and efficient review of the topical report. I think all of the feedback and discussions made sure that we had a complete product.

So I'm joined here by the lead technical contributor to the topical report, Dr. Matthew Denman, who will be giving the presentation today. We are also joined by several subject matter experts that will be available to answer detailed questions in their areas of expertise.

Just as a reminder to the Kairos attendees, if you do come off mute, please remember to introduce yourselves.

And with that I will turn it over to Matt. And let me make sure I can share my slides on this computer.

DR. DENMAN: Yes. So while Drew is pulling the slides up I'll begin my introductions.

Mr. Chairman, members of the Committee, thank you very much for your time today. My name is Matthew Denman and I am a principal reliability engineer at Kairos Power and it is going to be my pleasure today to brief you on Kairos Power's Mechanistic Source Term Methodology Topical Report.

And, Weidong, can you make sure that I'm a presenter so I can share my screen?

MR. WANG: I think you are. You are the presenter.

DR. DENMAN: The --

MR. WANG: Maybe you -- a different -- okay. Now it's because --

DR. DENMAN: Yes.

MR. WANG: -- you changed it up. Okay. Let me just go and make -- yes, it's changed.

(Pause.)

MR. WANG: We can see your screen now.

DR. DENMAN: Thank you so much.

Okay. So with that Kairos Powers' mission is to enable the world's transition to clean energy with the ultimate goal of dramatically improving people's quality of life while protecting the environment.

In order to achieve this mission we must prioritize that our efforts focus on a clean energy technology that is both affordable and safe. Today's topic, mechanistic source term, is key to allowing Kairos Power to demonstrate the safety of our design which will enable the affordability of that design.

At a high level our approach to source term is to decompose the problem into a series of material-at-risk throughout the plant and barrier release fractions that will separate that material at risk from our receptor at the site boundary.

For each barrier radionuclides are grouped and then we model the release of that group of radionuclides through the barrier using a representative element. Barriers for radionuclide release are the TRISO fuel and the FLiBe coolant. These form our functional containment for radionuclides.

Again radionuclide groupings are used to facilitate transport of radionuclides through the barriers and unique grouping structures will exist for various release models. So for say the fuel you might have a different grouping structure for mechanical grinding of the fuel verse a diffusion of radionuclides through the TRISO barriers.

At steady --

CHAIR PETTI: Matthew?

DR. DENMAN: Yes, sir.

CHAIR PETTI: I just have a real high-level question here on the methodology.

DR. DENMAN: Sure.

CHAIR PETTI: I understand it's to be used really for accidents, but do you guys plan to use this same methodology to support the worker dose evaluations, shielding needs, or are you guys thinking about a completely different approach there?

DR. DENMAN: That is a very good question and thank you very much for it. The approach in the topical is limited to off-site dose calculations and explicitly excludes worker does or control room dose. Similar methods may be used to quantify those dose metrics, but the complete strategy of how to drive a conservative consequence estimate has not been included in this topical report.

CHAIR PETTI: Okay. Thanks.

DR. DENMAN: So for sources of steady state material at risk in the system the overwhelming majority of our material at risk is contained within our TRISO fuel. That TRISO fuel can exist in multiple configurations. Most of our TRISO fuel will exist as either completely intact particles or with a compromised inner or outer PyC layer and all of these configurations are expected to retain an overwhelming majority of the fission products and heavy metals contained within those particles.

There will be -- due to manufacturing and in-service, steady state in-service failures there will be some TRISO particles that will have compromised silicon carbide layers, and these particles are expected to release a certain quantity of their fission products into the FLiBe coolant during steady state irradiation.

Additionally, as part of the manufacturing process there is a very small fraction of dispersed uranium that is expected throughout the fuel form and the fission products from this dispersed uranium have no credited fission product or heavy metal retention capabilities within the fuel.

These fission products and heavy metals will move into the circulating activity where they will be combined with impurities that are expected within the salt including sodium, uranium, thorium, various other corrosion products.

The circulating activity will continue to generate radionuclides via transmutation. There will be some tritium production within the FLiBe coolant.

The tritium primarily will either be absorbed into graphite pebbles and structural materials or will move into various off-gas cleanup systems.

When we talk about how we're going to quantify the material at risk throughout the plant, for the fuel we will focus on our manufacturing specifications. We will utilize the KP-BISON Fuel Performance Code to estimate the depletion of radionuclides from the fuel and we will use our Core Design Topical Report methodology in order to calculate the burnup and buildup of fission products within the fuel.

The circulating activity material at risk will be limited by technical specifications that will be set as limiting conditions of operations for our plant.

The holdup of tritium in structures and graphite pebbles will be calculated via the tritium source term methodology discussed in the next few slides. And various material at risks outside of our functional containment will be limited by its technical specifications, specifically the FLiBe cleanup and -- one sec.

(Pause.)

DR. DENMAN: Sorry. My apologies for

those technical difficulties.

The material at risk within the various cleanup systems will be limited by technical specifications.

For steady state tritium inventory evaluations, tritium will be -- or tritium modeling will include transport and holdup in the fuel pebbles and core moderator and graphite structures in the vessel and primary piping and intermediate heat exchange steel. Tritium is produced in the KP-FHR through the reactions listed below. The top two reactions are the primary reactions that contribute to tritium production in the system and the bottom two reactions are the primary reactions contributing to lithium-6 buildup, which will subsequently be sources of tritium production.

CHAIR PETTI: So, Matt, just another question. So you're not explicitly modeling tritium production in the graphite from lithium impurities nor ternary fission in the particles?

DR. DENMAN: So, well, we are --

CHAIR PETTI: I mean they may be significantly smaller here but --

DR. DENMAN: Yes.

CHAIR PETTI: -- it's probably worth just a -- I think they'll be smaller given the capability to make good graphite today. Years ago impurities were higher and you had to worry about those things.

DR. DENMAN: Understand. Yes, we are not explicitly modeling the lithium and the graphite, nor the ternary fission within the fuel due to the fact that an overwhelming majority of the tritium that is expected to be produced in the system will be produced via the FLiBe reactions shown on the slide.

CHAIR PETTI: Okay. Thanks.

DR. DENMAN: The KP-FHR is uniquely suited to retain radionuclides due to the large margins to fuel damage from our operating range. Our core inlet and outlet temperatures are in the 550 to 650 range. Our FLiBe freezing temperatures and our -- sorry, our FLiBe boiling temperatures are not until 1,430 degrees C. And our peak fuel temperatures above which we would potentially expect silicon carbide-induced failures aren't until 1,600 degrees C. So there is a large margin to the functional failure of our various radionuclide barriers within our functional containment approach.

For MAR mobilization for anticipated operational occurrences, design-basis events and design-basis accidents it should be emphasized that under design -- or under the conditions expected for these events only a minor fraction of the total material at risk in our plant can potentially be mobilized because a majority of our material at risk is contained safely within our TRISO fuel.

We expect no incremental fuel failures below 1,600 degrees C and there are multiple inherent safety features in our design to protect the fuel from achieving such high temperatures.

The material at risk in the reactor coolant as well as the material at risk presented in other locations can be mobilizing in anticipated operational occurrences, design-basis events, and design-basis accidents particularly via aerosolization of the FLiBe such as for jet breakup in a hypothetical guillotine rupture of a primary pipe or vaporization of chemical species within the FLiBe at elevated temperatures, although there are only limited release rates expected due to evaporation of soluble radionuclides from FLiBe at temperatures below 816 degrees C, which is our vessel limit and sets the upper bound of our design-basis --

CHAIR PETTI: Matt?

DR. DENMAN: Yes, sir.

CHAIR PETTI: Just the first sub-bullet in the first bullet, the way it's stated. The EPRI topical report has a failure fraction at -- what's called at high temperature. It's a statistical zero level because the testing showed there were no failures. Are you assuming that level for any accident event or are you saying zero is zero?

DR. DENMAN: We will use the KP-BISON Fuel Performance Topical Report to calculate the stresses and strains on the various barriers and the incremental fuel failure fraction. It is our expectation that that value will be near zero, below 1,600 degrees C, but our methodology is to actually calculate that.

CHAIR PETTI: So I have the same problem with the last topical. If that number is lower than what has been measured statistically, how do you validate that number?

DR. DENMAN: I will pass this question along to our fuel performance expert Ryan Latta.

Ryan, can you jump on?

MR. LATTA: Hello?

DR. DENMAN: Yes, Ryan?

MR. LATTA: Yes, this is Ryan Latta. Yes, the current methodology is to use the Fuel Performance Code. And it calculates the radiation history, uses the radiation history's input, and then goes through the transient analysis and uses the actual conditions that are for the accident, which are significantly below conditions that were tested for the furnace annealing test. So we probably have a 4 to 500 degree margin from the conditions that were in the furnace safety testing. So when you follow that track you end up with very low, near negligible failure fractions during an accident event. And so that's how the -- that's the methodology we followed for --

DR. DENMAN: And I will add -- this is Matthew Denman again. I will add that the methodology for determining where the -- or which barriers are intact are failed lies squarely within the fuel performance methodology. This topical report on source term basically looks only at -- once you've determined which barriers are available for release, how do you move radionuclides through those barriers?

So we kind of take the configuration of the TRISO fuel as a given boundary condition from the Fuel Performance Topical Report.

CHAIR PETTI: All right.

MEMBER REMPE: This is Joy. And first of all, I'd like to ask people who aren't speaking to mute their computers or phones because there's a lot of background noise I'm hearing, but a couple of questions.

I know the topical report says you don't -- now I'm getting an echo. So again, people, please mute. Okay?

But anyway, the topical report says you're not going to deal with beyond-design-basis events, but yet several times it talks about well, you'll just continue things for beyond-design-basis events. So could you clarify, are you planning to go ahead and use these same models and extend them for beyond-design-basis events or are you going to use a different methodology?

And then I didn't ask earlier, but I was curious, the topical report continues to say that, as other ones did, the coolant is an important barrier for release. And it doesn't talk about the fact that the coolant can interact with other barriers and degrade them. And how are you planning to modify this methodology to consider this degradation?

I'm sorry. Did -- I'm not hearing any response, so maybe now it's time to un-mute, whoever is trying to talk or respond.

MEMBER BLEY: Joy, I can hear you, so they ought to.

MEMBER REMPE: Thank you for that confirmation, but I was asking a lot of people to mute so maybe they haven't un-muted yet.

DR. DENMAN: Yes, I think I got muted without my knowledge. My apologies.

Thank you very much, Joy, for those questions. I will answer the beyond-design-basis question first.

So the methodologies that we developed for this topical report were the methodologies from phenomena that we expected to exist in anticipated operational occurrences, design-basis events, and design-basis accident boundary conditions. It is possible that in beyond-design-basis space that you will experience similar boundary conditions, and in those cases the methodologies may be extended into beyond-design-basis conditions. However, there are expected to be additional scenarios in beyond-design-basis event space that extend beyond the applicability of these models, and at that point we would have to revise and justify the models in a future license application. Does that answer your --

MEMBER REMPE: So you are planning to extend or do something with KP-BISON? You're not just going to say okay, now we're going to go and use something else that's similar to MELCOR or something like that, or you've just not decided yet what tool you'll use?

MR. PEEBLES: So this is Drew Peebles with Licensing. So not in this topical. So we would definitely deal with that in the future application of the methodology. So if we do extend beyond the design-basis, then we would have to justify how we're doing that in that future license application. But for this particular topical report we weren't asking for an NRC finding on beyond-design-basis conditions.

MEMBER REMPE: Again, I understand that you've said that you're limiting to design-basis events, but then in the report it continues to make reference to beyond-design-basis events and I'm not getting an answer to the question are you going to use this tool or another tool, or you've not decided what tool --

DR. DENMAN: Yes, I think the short answer is we haven't decided upon the beyond-design-basis --

(Simultaneous speaking.)

MEMBER REMPE: Okay. And then what about degradation due to long-term operation, from corrosion or something between the coolant, which you continue to say you think is a barrier, but one thing that's unusual about this design is that it's -- there's the potential that some of the barriers can degrade other barriers. And how are you planning to modify -- I didn't see anything discussed about how you will simulate that phenomena in this topical report.

DR. DENMAN: Thank you very much for that question, Joy. Particularly for the fuel in transient conditions we do not expect under very short time horizons for there to be induced failure of the fuel barrier such that you would have fuel/FLiBe interactions, and that is explicitly called out in the topical report.

Under longer term conditions if there were to be fuel/FLiBe interactions, then the radionuclides from the fuel would move into the FLiBe and join the circulating activity. And we have a technical specification on circulating activity, so as long as the circulating activity remains below that technical specification, our methodology would still hold.

MEMBER REMPE: So --

DR. DENMAN: Matthew. My apologies.

MEMBER REMPE: So you're basically saying you don't model degradation with long-term operation. You just think it's not going to be that important as long as the coolant circulating activity stays below a certain value?

DR. DENMAN: Correct.

MEMBER REMPE: That you're just not simulating that phenomena? Then what about can the circulating activity, if it were to start degrading other subsequent barriers due to corrosion of some of the structural material? Are you still -- are you also going to be neglecting it? And then you'll -- is this something that's built into the model, you constantly do a check to make sure the circulating activity stays below that value all the time so that you don't ever exceed this? So is that something you've put into KP-BISON to do some sort of check?

DR. DENMAN: So the circulating activity technical specification will be a limiting condition of operation. We will monitor the circulating activity over the life time of the reactor operations and ensure that we are below the value set forth in our license.

MEMBER REMPE: We're talking about the tool today. And so you're telling me well okay, so the tool doesn't have to consider this degradation interaction between the coolant barrier and the barriers within the fuel. So basically you're saying that if you're doing a simulation to provide some sort of source term to the NRC, you're constantly doing some sort of check to make sure that you don't -- if you're going to do source term after long-term operation, end of cycle, that you've done a check always in the tool to make sure it's below that value, right? Is what you're telling me?

DR. DENMAN: Thank you very much. Not quite. We will set a technical specification in our license application that sets the upper limit of circulating activity in our FLiBe. We will use KP-BISON to model normal buildup and diffusion of radionuclides out of the fuel, but that only sets the initial condition of material at risk within the fuel itself.

In the FLiBe for any accident condition we will use the technical specification -- or any design-basis accident condition we will use the technical specification value as the initial condition of circulating activity in the FLiBe. So we will not be calculating in an a priori estimating what that release would be. We will use the upper bound value of acceptable circulating activity as our initial condition for the accident.

MEMBER REMPE: Okay. And then what about if there's interactions between the FLiBe and structural material?

DR. DENMAN: So the Structures Topical Report will ensure that the vessel is not degraded with -- by the FLiBe. Within our anticipated operational range any other structure system and component is not safety-related, and breaks in those systems would be evaluated in our postulated event analysis.

MEMBER REMPE: Okay. Thank you.

DR. DENMAN: So as a part of this analysis we're not explicitly modeling that.

MEMBER REMPE: Okay. Thank you.

CHAIR PETTI: Just a clarification. So the tech spec on circulating activity, is that basically equivalent to what the gas reactor guys are talking about SARDL?

DR. DENMAN: They're related concepts, although we do not believe that we would set a limit on the circulating activity that would be the break point between acceptable or unacceptable off-site doses. We would choose a value that is -- that we believe is achievable to be monitored and measurable and ensure the safety of the system. But it might be slightly -- formulated in a slightly different way that the SARDLs.

CHAIR PETTI: Okay.

DR. DENMAN: Okay. So and then there's also going to be tritium that's going to be stored in the graphite pebbles and structures that can be desorbed at elevated temperatures and our methodology will examine that release.

Our design-basis accident site boundary dose will be used -- dose is going to demonstrate that the KP-FHR meets dose limits in 10 C.F.R. 50.34, 52.79, and 100.11. Again, technical specifications will be set on the activity of the FLiBe, cover gas and other systems, and the system will be design to preclude incremental fuel failures from DBA conditions as evaluated by KP-BISON.

Anticipated operational occurrences and design-basis event source term analyses are similar to design-basis accidents, but more realistic assessments of barriers, mitigation strategies and initial conditions may be assumed.

The circulating activity technical specification will be used to inform operational limits on circulating activity. These operational limits may be more realistic conditions for normal operation effluent calculations as well as anticipated operational occurrences and design-basis events.

Our radionuclide grouping and transport approach is very similar to that used in light water reactor safety analysis. I have the MELCOR grouping structure on the right here where you can see the various chemical groups and then the representative element at the top that represents now the releases from those groups are calculated for light water reactor. We take a similar approach, although we evaluate the grouping structures specifically to the barrier and the release mode within that barrier.

So essentially our approach is we look at individual isotopes within a barrier and combine them into their RN groups, their radionuclide groups. We calculate the release fractions for each radionuclide group associated with the medium as calculated by driving forces within that barrier: temperatures, pressures. Release fractions are combined with relevant inventories to determine the quantity of that material that is mobilized.

Once you move from one barrier to the next the radionuclide inventory is combined with any radionuclides that are already present in that next barrier and then regrouped for subsequent mobilization. Once you reach the gas space the dose consequences for radionuclides that are transferred into the gas space are evaluated with RADTRAD and ARCON.

CHAIR PETTI: So, Matthew, I had bad network quality there for a minute so I missed it, but the groupings are the same no matter where the fission product is in the system, or does it -- when it's in the fuel it's considered one way because of the chemistry there. When it's in the salt it's considered because of the chemistry there?

DR. DENMAN: Correct. Every barrier will have its unique grouping structure. And specifically for the fuel there is a unique grouping structure for diffusion versus mechanical grinding of the fuel. So different release pathways may have their own unique grouping structure compared to the -- and then each barrier will have its own unique grouping structure.

CHAIR PETTI: Okay.

DR. DENMAN: Okay. Our primary barrier for radionuclide retention is our TRISO fuel. This fuel contains an overwhelming majority of the material at risk within our plant during normal and off-normal operating modes. Again, a series of diverse and robust barriers to radionuclide retention with extensive industrial fabrication experience and irradiation under a variety of conditions such as the

Advanced Gas Reactor Development Program as mentioned earlier.

Our TRISO fuel manufacturing specifications will determine how the fuel configurations begin in the transient. Fission products will diffuse from imperfect particles, primarily particles with failed silicon carbon layers or that have exposed kernels. Radionuclides from heavy metal contamination from the manufacturing process will have no credit for radionuclide retention in steady state.

Minimum expected steady state diffusion of radionuclides are expected from the remaining configurations with intact silicon carbide layers. Compromised configurations will partially or entirely be depleted fission products during steady state thus reducing the available material at risk within those TRISO configurations during the transient.

For the FLiBe barrier this is the second part of our functional containment for radionuclide retention. Once radionuclides are in the FLiBe they will be separated into either salt soluble compounds, suspended oxides, noble metals or gases. And Kairos Power's Fuel Development Program, or sorry, FLiBe Development Program builds on radionuclide retention experience in the molten salt reactor experiment with the exception that our salt is going to be much, much cleaner than what was experienced in MSRE, which was a fuel salt system.

CHAIR PETTI: So, Matthew, just a question on that, and if I get into proprietary stuff, just tell me and we'll cover it in the closed session.

I noticed that you had put some fissile impurities in the salt, and I was surprised at that level being that high. And I wasn't sure if that was just being conservative or what was done back in the old days of MSRE or whether that actually is what you get.

My experience in gas reactors is in the old days stuff was just not as clean as you can get today with today's technology and I wasn't sure whether this was a holdover from that. I would have thought you'd probably be able to get better, cleaner salt than that.

DR. DENMAN: So the cleanest of the salt is going to be dependent upon the economics of the system and how much we want to pay for various grades. Those decisions are not made at this point in time and our methodology is designed to be flexible enough to account for various levels of impurities.

CHAIR PETTI: So that's sort of a -- let's just say a conservative level. That may not be what you actually see in practice.

DR. DENMAN: Correct.

CHAIR PETTI: Okay.

DR. DENMAN: For tritium transport, tritium transport within structures is determined by mass transfer coefficients from FLiBe flow characteristics throughout the system. Transport within structures is determined based upon material properties such as diffusion within and through steel, diffusion and trapping within pebbles and structural graphite.

Salt structure boundary conditions set by material tritium -- is set by material tritium solubility, particularly Henry's Law for solubility of tritium fluoride and tritium gas in FLiBe, and Sievert's Law for solubility of tritium in steel.

CHAIR PETTI: So, Matthew, just a comment here. The amount of literature on tritium behavior in these materials, both the salt and the graphitic material, is quite large and there's a lot of uncertainty. These measurements are not easy to make. Diffusion in liquids are notoriously difficult and have high uncertainty. Solubilities are not easy.

It is now understood better that simple experiments where one injects tritium into molten coolants like this show a bias. The tritium doesn't actually go in. It can sit along the surface. If you think of like a loop. This has been shown in Europe in the Fusion Program for a different coolant that's a low-solubility coolant. FLiBe is a low-solubility coolant.

And I think it's very difficult. This is exactly how I would model it. I just think the validation is going to be quite challenging because the experiments may have these biases that you -- until you get to the actual in situ generation of tritium, you may be surprised. And it's just something that I think when you're doing sensitivity studies on the model you got to open up the window here because there's a lot of stuff that even though the experimentalists have done the best that they can do, without in situ generating tritium it's really difficult.

In terms of the graphitic material I would again caution that the fusion experiences on graphites that are not these graphites. Pebble graphitic material is not a graphite, whereas the -- your reflective material is a nuclear graphite. Those are very different microstructures so that there's very big differences potentially in trapped concentrations. I believe the trapped energies are probably generic to carbon materials, but the actual concentrations are very strongly microstructural-dependent. Radiation can affect it, too. All of these things make it much more complicated than these really nice elegant models.

And if you go back -- you have to go back a little ways in the fusion world to see some of the models and the differences and some of the complexity there. It's just a caution that when you think about the validation, you think about sensitivity, keep the window open large because of these differences.

I also recommend that if you haven't looked at complexity of models, take a look at the modeling that's done to date. There has been recent publications on air ingress with graphite. Oak Ridge and Idaho have done a tremendous amount of modeling, highly complex, and they try to bring in the microstructure. And it takes you back to say, wow, there's a lot there. They spent a decade getting all the parameters that you need to really understand it.

And then you look at these models which are much simpler and it just gives you a cause for concern. So it's worth looking at some of that as you think about how you're going to bound things and do sensitivity analysis and the like.

DR. DENMAN: Thank you very much for your feedback on that. It's very valuable and insightful and we'll take it as we move forward with this approach.

For tritium from the FLiBe-free surface, tritium fluoride and tritium gas can both exist as dissolved gases in the FLiBe. Contributions to off-site dose would either require permeation into vessel or piping and then release into the reactor building or evolution into the gas space which is modeled via the gas transport equations influenced by the experiments as shown below.

For gas space analysis we are using the NRC codes RADTRAD and ARCON96. These are used to model radionuclides traveling through the building and off site. And to support dose calculations the existing models and framework set forth in these codes are accepted as is.

For RADTRAD as input we need the mobilized material at risk from the previous barriers as previously discussed. RADTRAD will handle all the radionuclide decay and for the entire duration of the transient and use the Henry correlation for aerosol settling, and conservatively and prescriptively leakage rates will be applied out of the reactor building into the environment.

For ARCON the release definitions around the release of radioactive material from the site the location of the receptor and meteorological conditions at the site are needed to calculate chi over qs.

Various limitations are set forth in this topical report. They are listed on the slide, but I will not read them word for word.

And with that are there any further questions?

CHAIR PETTI: Just another comment in the tritium realm with the nitrate salt. Whenever one is dealing with lower levels of tritium, there's always a waste management concern. You get to a point where the concentrations are so low it's hard to find a disposal route. The folks in EDA (phonetic) have been struggling with this. When you have lots of tritium, there's lots of technologies to be able to concentrate it, move it, get it where you want it to be, put it on a bed or something, but when you get to low concentrations, it's above what's allowed to be released, but it's so low that the technologies to deal with it are a problem. So it's just something to put on your tickler list as your design evolves.

Again my information may be a little out of date, but this was at least the case ten years ago, but they were still struggling with some of these sorts of issues.

DR. DENMAN: Thank you very much for that feedback. It's definitely something that we'll take back as we continue to mature our design.

I'm not able to see the chat window or anything, so if there's any further questions?

CHAIR PETTI: Yes, members, any questions?

DR. DENMAN: Well, hearing none, I really appreciate your time in this open session and look forward to continued conversations in the closed session.

CHAIR PETTI: Okay. Thanks.

Is Michelle going to talk? Who's going to talk for the staff?

MR. CUADRADO DE JESUS: For the staff we don't have presentations for the open session.

CHAIR PETTI: Ah, okay. Then I guess with that we can move to the closed session.

MR. WANG: Dave?

CHAIR PETTI: Yes.

MR. WANG: We need to have public comment.

CHAIR PETTI: Yes, yes, yes. So okay, let's open -- anybody that has a comment from the public, *6 to un-mute yourself. Give us your name and your comment.

Okay. Hearing none, I guess we will end this open session. And I think all the members should have the link to the closed session.

And, Kairos, we'll want you to make sure that all the folks you think should be there should be there and Weidong and our staff will handle the NRC side.

So with that we'll see everybody in the closed session.

(Whereupon, the above-entitled matter went off the record at 9:56 a.m.)

Enclosure 2

Open Session Presentation Slides for the November 19, 2021
ACRS Kairos Power Subcommittee Briefing

(Non-Proprietary)

Kairos Power

KP-FHR Mechanistic Source Term Methodology Topical Report

ACRS Subcommittee Meeting, November 19, 2021

Kairos Power’s mission is to enable the world’s transition to clean energy, with the ultimate goal of dramatically improving people’s quality of life while protecting the environment.

In order to achieve this mission, we must prioritize our efforts to focus on a clean energy technology that is *affordable* and *safe*.

High Level Approach

Source Term Methodology

- Decompose the problem into a series of Material at Risk (MAR) and barrier Release Fractions (RFs) that separate that MAR from a receptor at the site boundary.
- For each barrier, group radionuclides into and model release through that barrier using a representative element for that group.
  - The barriers for radionuclide release are the TRISO fuel and the Flibe coolant (i.e., functional containment).
  - Radionuclide groups are used to facilitate transport through barriers.
  - Unique grouping structures exist for specific release modes (e.g., mechanical grinding of fuel in the PHSS vs diffusion through TRISO barriers).

$$ST^{i}(t) = \sum_{j=1}^{J} MAR_{j}^{i}(t) \prod_{j}^{J} RF_{j}^{i}(t)$$

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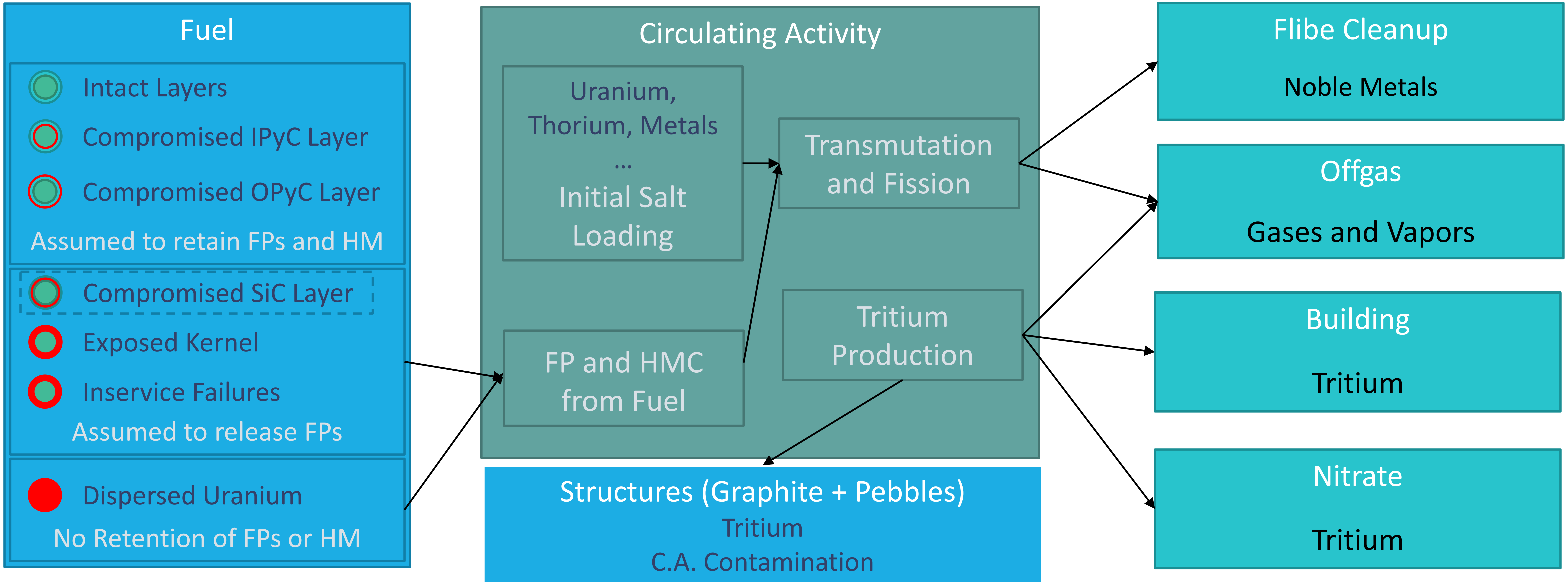
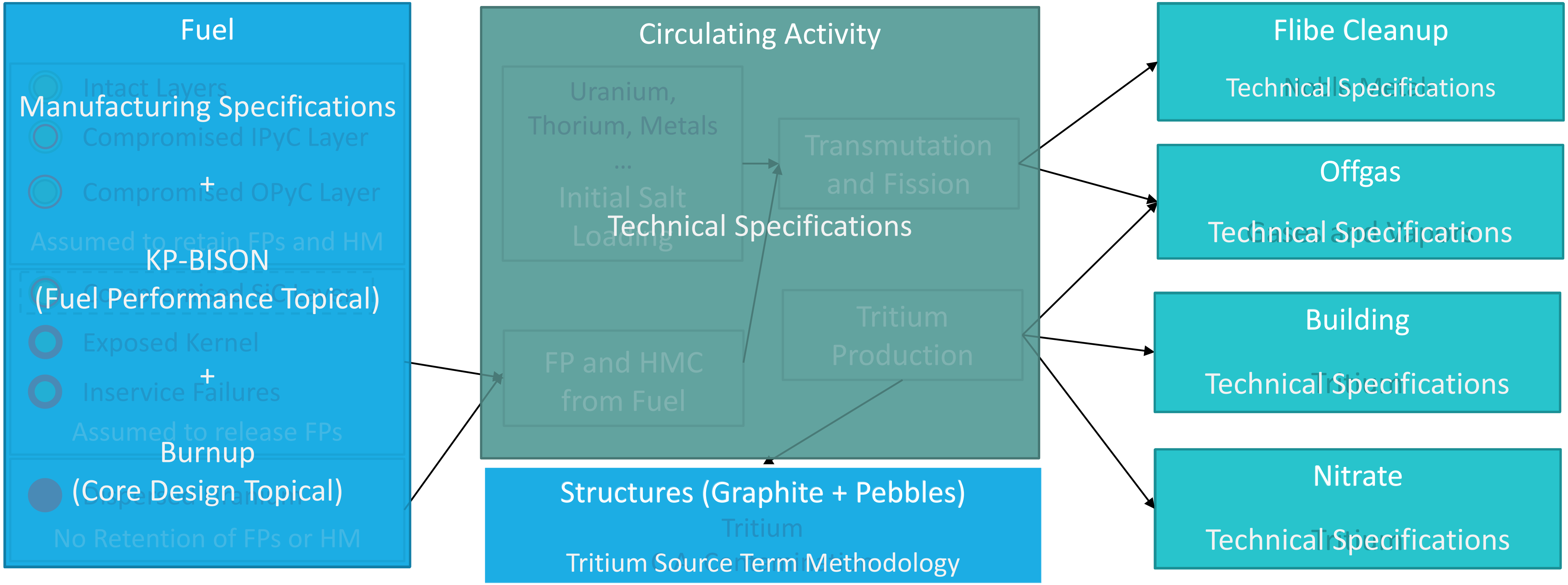
Sources of Steady State Material at Risk (MAR)



Sources of Steady State Material at Risk (MAR)



Steady State

Tritium Inventory

- **Tritium modeling will include transport and holdup in:**
  - **Fuel Pebbles and core moderator**
  - **Graphite Structures**
  - **Vessel Steel**
  - **Primary piping**
  - **Intermediate Heat Exchangers**

- **Tritium is produced in the KP-FHR through the following reactions:**

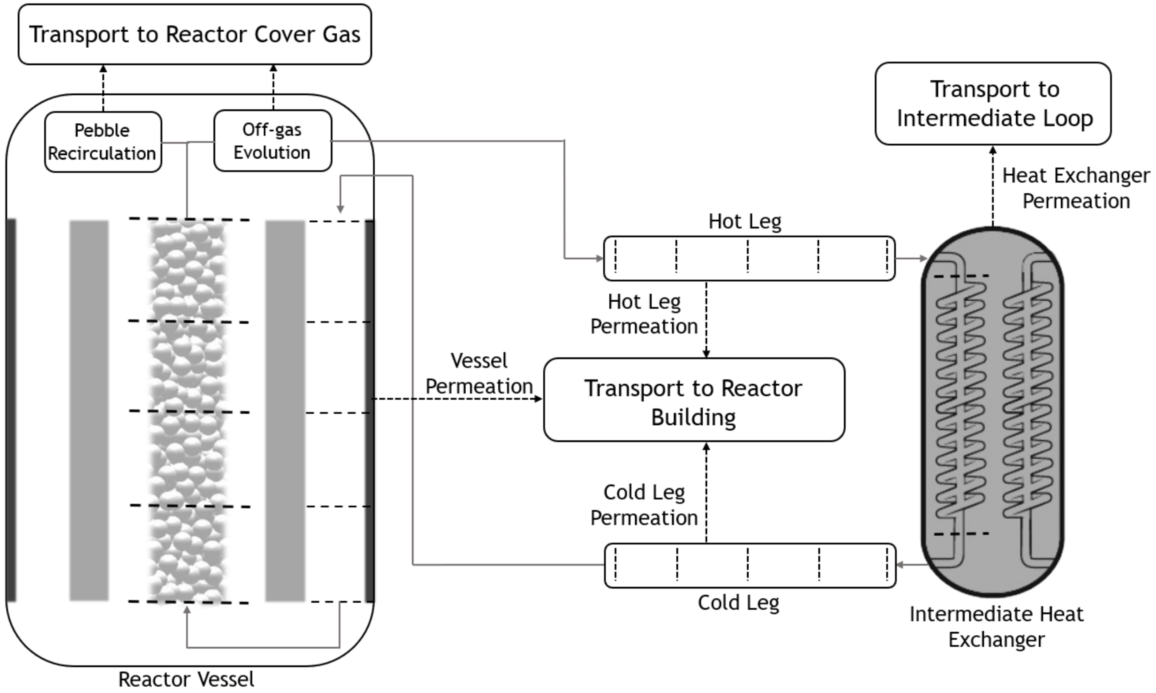
$$^{6}Li + n \rightarrow {}^{4}He + {}^{3}H$$
$$^{7}Li + n \rightarrow {}^{4}He + {}^{3}H + n'$$

Modeled Tritium Production Reactions

$$^{9}Be + n \rightarrow {}^{7}Li + {}^{3}H$$
$$^{19}F + n \rightarrow {}^{17}O + {}^{3}H$$

$$^{9}Be + n \rightarrow {}^{6}He + {}^{4}He$$
$$^{6}He \rightarrow {}^{6}Li + \beta + \bar{\nu_e} \quad (\mathrm{t}_{1/2} = 0.8\ \mathrm{s})$$

Modeled Li-6 build-in Reactions



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KP-FHR Specifications

Uniquely Large Margins Between Operational and Failure Temperatures

Parameter	Value/Description
Reactor Type	Fluoride-salt cooled, high temperature reactor (FHR)
Core Configuration	Pebble bed core, graphite moderator/reflector, and enriched Flibe molten salt coolant
Core Inlet and Exit Temperature	550°C / 600-650°C

Design Temperature Limits	Value
Primary Salt (Flibe) Freezing and Boiling Temperatures	459°C / 1430°C
Maximum ASME Section III, Division 5, SS316 Temperature	816°C
Peak Fuel Temperature Limit	1600°C

Our combination of fuel and coolant provides a uniquely large safety margin.

MAR Mobilization in AOOs, DBEs, and DBAs

Only minor fractions of the total MAR can be mobilized in AOOs, DBEs, or DBAs

- The vast majority of MAR is safely protected in the fuel during AOOs, DBEs, and DBAs.
  - No incremental fuel failure is expected at temperatures <1600°C.
  - Multiple inherent safety features protect the fuel from achieving high temperatures.
- MAR circulating in the reactor coolant as well as MAR present in other locations (cover gas, intermediate loop, etc) can be mobilized in AOOs, DBEs, and DBAs.
  - Aerosolization of Flibe – Hypothetical guillotine pipe break or primary pump operations
  - Vaporization– chemical specific evaporation is evaluated across accident temperature profiles
    - Limited release rates are expected from evaporation of soluble radionuclides from Flibe for temperatures below 816°C.
- Tritium stored in graphite, pebbles, and structures can be desorbed at elevated temperatures.

AOO, DBE, & DBA Source Term Methodology

- DBA site boundary dose to demonstrate KP-FHR meets dose limits in 10 CFR 50.34, 10 CFR 52.79, and 10 CFR 100.11.
- A technical specification (tech spec) limit will be set on activity in the Flibe, cover gas, and other systems.
  - The system is designed to preclude incremental fuel failures due to the DBA conditions as evaluated by KP-BISON.
- AOO and DBE source term analyses similar to DBAs, but a more realistic assessment of barriers, mitigation strategies, and initial conditions may be assumed.
- The circulating activity technical spec. will be used to inform an operational limit on circulating activity. This operational limit can be used as a more realistic initial condition for normal operation effluent calculations as well as certain AOOs and DBEs.

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Radionuclide Grouping and Transport Approach

- Transport of radionuclides through each medium is evaluated on an RN group basis using the following steps:
  1. Individual isotopes are combined into RN group for each barrier.
  2. Release fractions of each RN group associated with that medium is calculated given driving forces (e.g., temperature, pressure).
  3. Release fractions are combined with the relevant inventories to determine the quantity of material that is mobilized. That incoming material is then:
     1. Combined with the radionuclides already present in the next barrier and then
     2. Regrouped for subsequent mobilization
  4. The dose consequences for radionuclides that are transferred into the gas space are evaluated with RADTRAD and ARCON.

LWR Example

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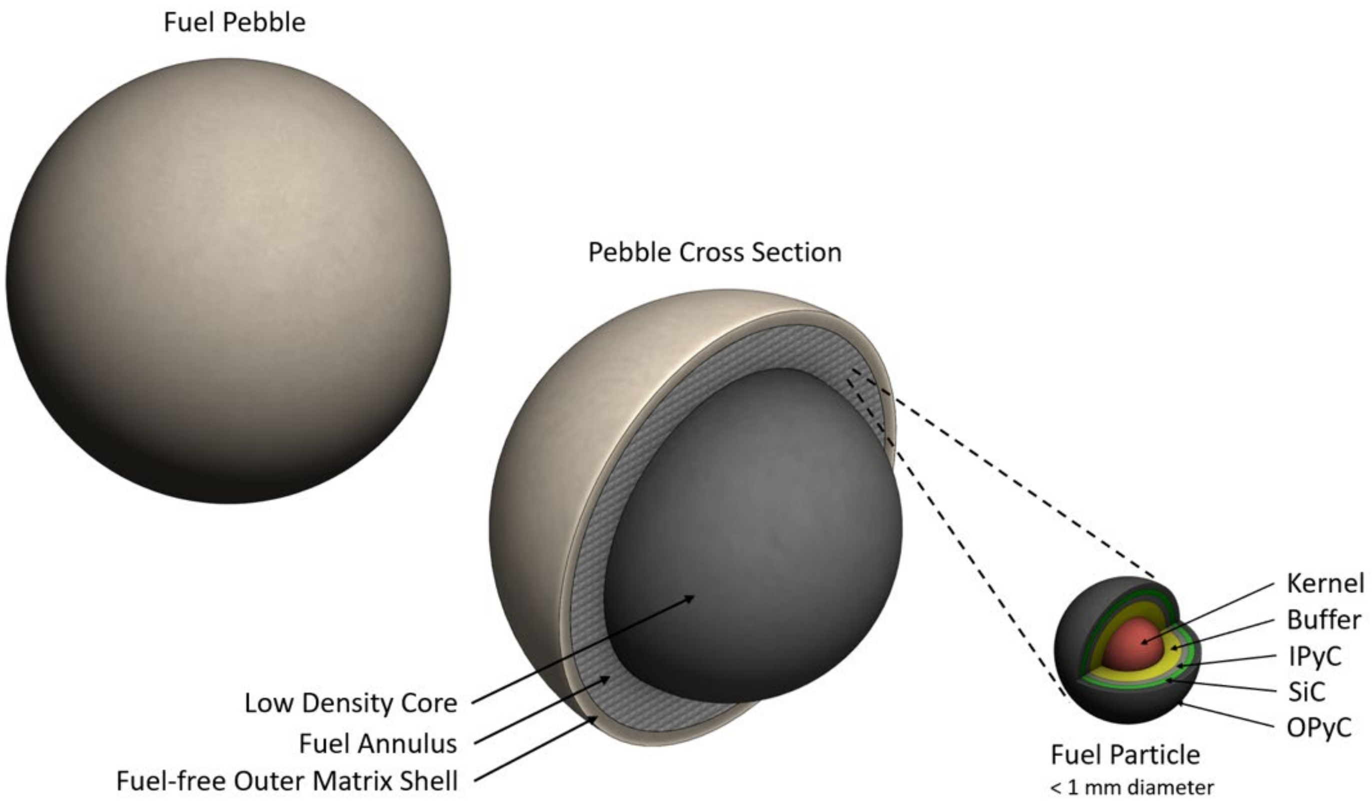
KP-FHR Fuel Element

The Primary Barrier of Radionuclide Retention

- The TRISO fuel form provides the first barrier to radionuclide retention in the KP-FHR for all normal and off-normal operating modes
- TRISO particles utilize a series of diverse barriers to provide robust fuel performance
- Kairos Power's fuel design builds on the AGR fuel development program
  - Extensive industrial fabrication experience
  - Validated irradiation performance under a wide variety of conditions



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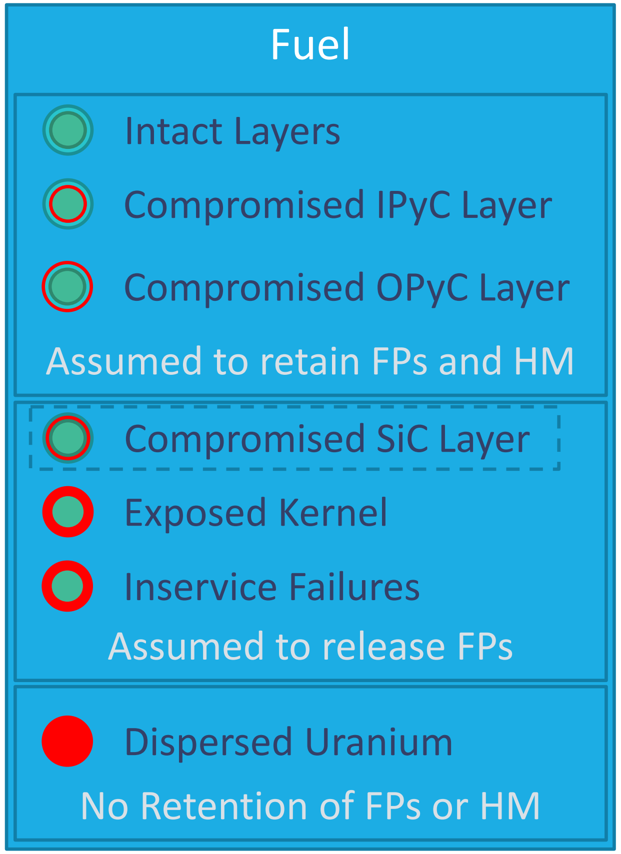
TRISO Fuel

Configurations for Material at Risk (MAR)

- Fuel manufacturing specifications will determine what radionuclides begin the transient in the fuel:
  - Fission products (FPs) diffusion from imperfect particles
    - Compromised SiC layer but intact IPyC and/or OPyC will release a smaller a range of FPs to the Flibe.
    - Exposed kernels and in-service TRISO failures mobilize mobile a larger range FPs transported to the Flibe at steady state.
    - Radionuclides from heavy metal contamination from the manufacturing process will have no credited retention at steady state.
  - Minimal expected steady state diffusion of radionuclides are expected from the remaining configurations.
- The compromised configurations will be partially or entirely depleted of FPs during steady state, thus reducing the MAR available for release during the transient.



TRISO Cohorts

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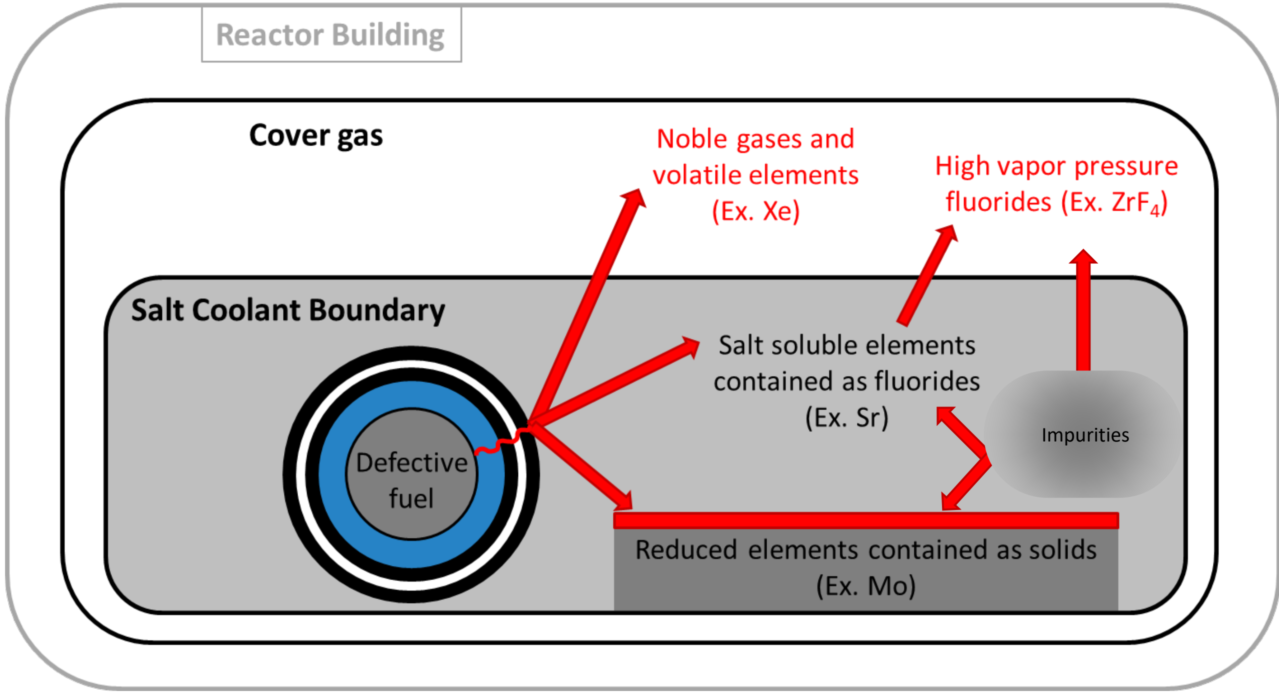
KP-FHR Flibe Coolant

The Second Barrier of Radionuclide Retention

- The primary coolant, Flibe, provides a secondary functional containment barrier to radionuclide retention in the KP-FHR for all in-core normal and off-normal operating modes.
- Flibe can chemically react with fission and activation products, separating them into:
  - salt soluble compounds
  - suspended oxides
  - noble metals, or
  - gas phases
- Kairos Power's Flibe development program builds on radionuclide retention experience in the Molten Salt Reactor Experiment

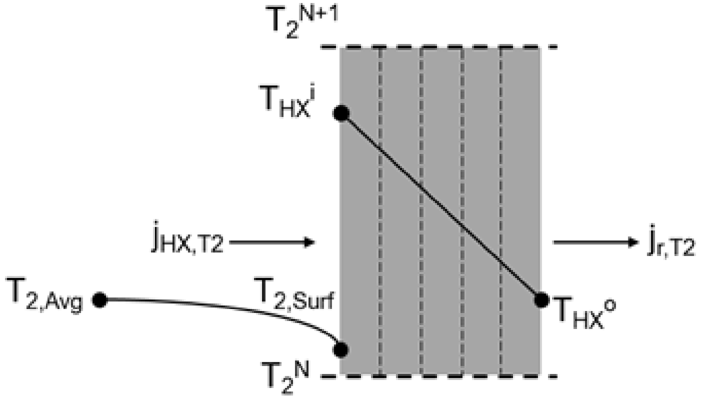


Tritium Uptake into Structures

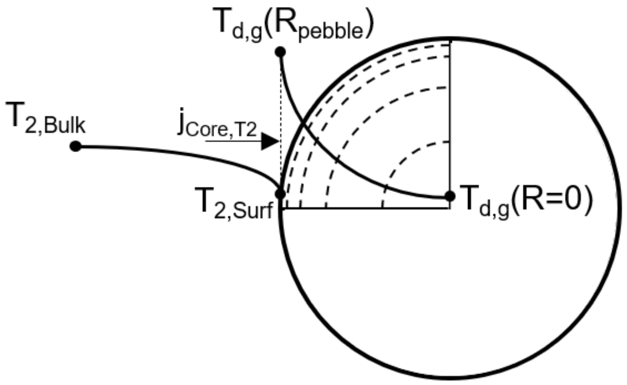
Metallics, Fuel and Moderator Pebbles, Structural Graphite

- Tritium transport to structures determined by mass transfer coefficients from Flibe flow characteristics
- Transport within structures determined based on material properties
  - Tritium diffusion modeled within steels
  - Tritium diffusion + trapping modeled within pebbles and structural graphite
- Salt/Structure boundary condition set by material tritium solubility
  - Henry's law solubility for TF and T_2 in Flibe
  - Sievert's law solubility for T in steel

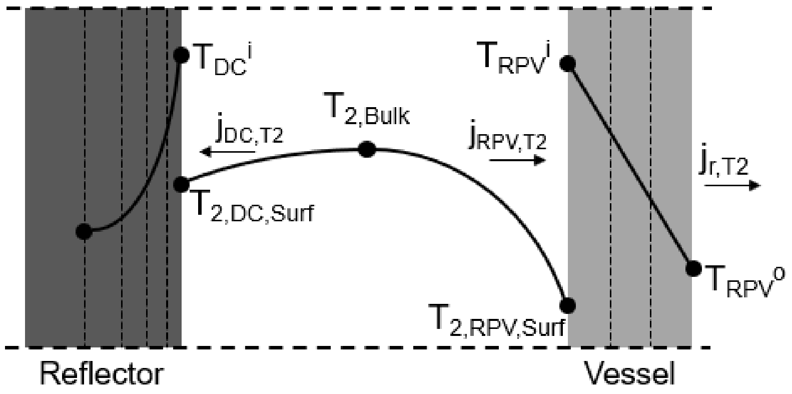
Examples:



Heat Exchanger – Permeation through Metal



Core – Retention in Pebbles



Downcomer – Vessel Permeation and Reflector Retention

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Tritium from Flibe Free Surface

Evolution into the Cover Gas

- Tritium fluoride (TF) and Tritium (T_2) can both exist as dissolved gases in Flibe.
- Contribution to offsite dose would require either:
  - Permeation into vessel or piping and then releasing to the reactor building or
  - Evolution into the cover gas which is modeled using mass transport equations influenced by experiments conducted by Suzuki et al.

$$T_2\ Evolution\ \left[\frac{mol}{s}\right] = k_{Evol,T_2} A_{Interface} \left(T_{2,Salt} - T_{2,gas}\right) \cong k_{Evol,T_2} A_{Interface} T_{2,Salt}$$

$$TF\ Evolution\ \left[\frac{mol}{s}\right] \cong k_{Evol,T_2} A_{Interface} TF_{Salt} \left(D_{TF}/D_{T_2}\right)$$

A. Suzuki, T. Terai and S. Tanaka, "Tritium release behavior from Li2BeF4 molten salt by permeation through structural materials," *Fusion Engineering and Design,* Vols. 51-52, pp. 863-868, 2000.

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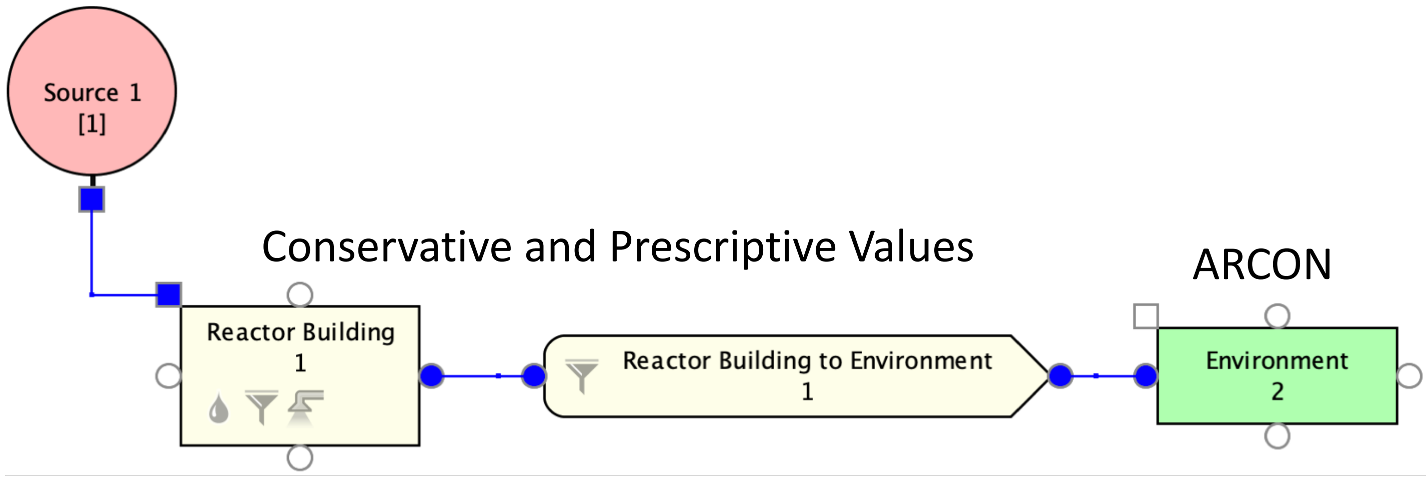
Gas Space Analysis

Simple and Conservative

- Codes: RADTRAD and ARCON96
  - Gas space transport
  - Dose calculations
  - Existing models are accepted as-is.
- Key Inputs:
  - RADTRAD:
    - Mobilized material-at-risk activities
    - Depletion mechanisms
      - Radioactive decay and/or
      - Henry correlation for aerosol settling.
    - Leakage rates (Conservative)
  - ARCON:
    - Release definitions
    - Receptor definitions
    - Meteorological data

KP MST Models



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Limitations

- 1. Approval of KP-Bison for use in fuel performance analysis as captured in KP-TR-010-P (KP-FHR Fuel Performance Methodology).
- 2. Justification of thermodynamic data and associated vapor pressure correlations of representative species.
- 3. Validation of tritium transport modeling methodology.
- 4. Confirmation of minimal ingress of Flibe into pebble matrix carbon under normal and accident conditions, such that incremental damage to TRISO particles due to chemical interaction does not occur as captured in KP-TR-010-P (Fuel Qualification Methodology for the KP-FHR).
- 5. Establishment of operating limitations on maximum circulating activity and concentrations relative to solubility limits in the reactor coolant, intermediate coolant, cover gas, and radwaste systems that are consistent with the initial condition assumptions in the safety analysis report.
- 6. Quantification of the transport of tritium in nitrate salt and between nitrate salt and the cover gas
- 7. The phenomena associated with radionuclide retention discussed in this report is restricted to molten Flibe. The retention of radionuclides in solid Flibe is beyond the scope of the current analysis.
- 8. The methodology presented in this report is based on design features of a KP-FHR (details provided in report). Deviations from these design features will be justified by an applicant in safety analysis reports associated with license application submittals.