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1716 14th Street, N.W.
Washington, D.C. 20009
(202) 234-4433

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

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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704TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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OPEN SESSION

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THURSDAY

APRIL 6, 2023

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The Advisory Committee met via hybrid In-Person and Video-Teleconference, at 8:30 a.m. EDT, Joy L. Rempe, Chairman, presiding.

COMMITTEE MEMBERS:

JOY L. REMPE, Chairman
WALTER L. KIRCHNER, Vice Chairman
DAVID A. PETTI, Member-at-Large
RONALD G. BALLINGER, Member
CHARLES H. BROWN, JR., Member
VICKI M. BIER, Member
VESNA B. DIMITRIJEVIC, Member
GREGORY H. HALNON, Member

JOSE MARCH-LEUBA, Member
MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:
DENNIS BLEY
STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:
CHRISTINA ANTONESCU
LARRY BURKHART

ALSO PRESENT:
JOE ASHCRAFT, NRR
ERIC BENNER, NRR
GILBERTO BLAS RODRIGUEZ, NRR
SAMIR DARBALI, NRR
WILLIAM JESSUP, NRR
CHRIS LEVESQUE, TerraPower
ED LYMAN, Public Participant
KHOI NGUYEN, NRR
TARA NEIDER, TerraPower
JASON PAIGE, NRR
RYAN SPRENGEL, TerraPower
RICHARD STATTEL, NRR
DINESH TANEJA, NRR

MARK WERNER, TerraPower

ERIC WILLIAMS, TerraPower

GEORGE WILSON, TerraPower

BRIAN YIP, NSIR

P R O C E E D I N G S

8:30 a.m.

CHAIR REMPE: Good morning. It is 8:30 on the East Coast, and this meeting will now come to order. This is the second day of the 704th Meeting of the Advisory Committee on Reactor Safeguards.

I'm Joy Rempe, Chairman of the ACRS. Other members in attendance are Ron Ballinger, Vicki Bier, Charles Brown, Vesna Dimitrijevic, Greg Halnon, Walt Kirchner, Jose March-Leuba, Dave Petti, and Matt Sunseri. We do have a quorum.

Similar to yesterday, the Committee is meeting in person and virtually. A communications channel has been opened to allow members of the public to monitor the Committee discussion. Mr. Larry Burkhart is the Designated Federal Officer for today's meeting.

During today's meeting, the Committee will consider the following topics: planning and procedures session and Commission meeting preparation, TerraPower and Natrium reactor design overview and 3D model walkthrough. I note that portions of our discussions on both of these items may be closed.

The transcript of the open portions of the discussion on topic two is being kept, and it's requested that speakers identify themselves and speak with sufficient clarity and volume so they can be readily heard. Additionally, participants should mute themselves when they're not speaking.

At this time, I'd like to ask other members if they have any opening remarks. Not hearing any, then I'd like to go off the record at this time before we start with topic one today. And I'd also ask the court reporter to return at 1 p.m. for our discussion on topic two.

(Whereupon, the above-entitled matter went off the record at 8:31 a.m. and then went back on the record at 1:00 p.m.)

CHAIR REMPE: Okay. It's 1 p.m. on the East Coast, and we are going to reconvene, and I'm going to ask Member Kirchner to lead us through the second topic for today. Walt.

VICE CHAIR KIRCHNER: Okay. We have guests here today with us from TerraPower, and they are going to give us an informational presentation on the Natrium design they are developing and give us an overview. I'd ask Chris Levesque to perhaps somewhere through this address the status of their engagement with the staff, which is useful information for us when we look at our longer-range planning going forward.

So with that, Chris, I'm just going to turn it over to you to introduce your team. And thank you for being here. We like the engagement in person, so, hopefully, you'll like the engagement with us in person, as well. Go ahead.

MR. LEVESQUE: Thank you, Madam Chair, Mr. Vice Chair, and thanks to the entire Advisory Committee on Reactor Safeguards at this time. We look forward to the engagement today and to multiple engagements with you in the future as we go forward with the Natrium licensing process.

I'm joined by Tara Neider, who is the project director for the Natrium project, Senior Vice President and project director at TerraPower. She oversees the entire Natrium project, including the 800 design engineers who are working on the project today in the procurement and instruction processes, as well.

I think I might defer further intros from the TerraPower team because we'll have multiple people coming up today, and we'll save those intros for when those subject matter experts come in.

I wanted to begin by sharing, you know, the common value, the common understanding that we at TerraPower feel towards reactor safety. We believe the Natrium design, you know, addresses the NRC's requirements and the engineering requirements that are needed for reactor safety. It is a Gen IV reactor, which we expect Gen IV reactors to have even higher margins for safety and we'll utilize excessively safe and inherent systems to achieve safety.

MR. NGUYEN: Excuse me. Could you identify yourself for the court reporter? Thank you.

MR. LEVESQUE: Yes. My name is Chris Levesque, President and CEO of TerraPower.

We're also looking forward, in addition to the engagement with the ACRS, we're closely monitoring the development of the Advanced Reactor Content of Application Project. We know, following the Nuclear Energy Innovation and Modernization Act, there's various developments going on in regulation, so we're watching these closely. But we are making this first application under 10 CFR 50, an established statute.

We're also following the NRC's guidance on the pre-application engagement, and we have considerable pre-application engagement already behind us with 39 meetings in pre-application, 5 topical reports, and 3 white papers. And that engagement with the NRC has been very constructive. We look forward to engagements like this with the ACRS and engagements with the NRC in the pre-application review as a way of improving our eventual construction permit application and operating license application.

Knowing that Natrium is a Gen IV design and most of the experience in the U.S. is in the light water arena, we're doing as much as we can to familiarize the staff and other stakeholders on sodium fast reactors. Best example of that is the multiple training sessions we've given for NRC staff on sodium fast reactors and the attributes of Natrium.

I want to finish by noting that this first Natrium plant, which will be built in the state of Wyoming at a retiring coal plant, is being built in conjunction with a very important public-private partnership, the Advanced Reactor Demonstration Program. Both TerraPower and X-energy were recipients of the ARDP Demonstration Project Award. That award involves the Department of Energy funding half of the costs of these first projects, and that's intended to help us overcome many first-time costs associated with the first license, the first design, learning curve, frankly, because the U.S. has somewhat fallen out of experience with new builds.

So the ARDP, we feel, is a very, very important national program. We're proud to be recipients of that program, and a very important part of that program is that the first owner of the Natrium nuclear power plant will be a commercial owner.

So we believe that public-private partnerships are very important. We're competing on the world stage with Chinese and Russian-state owned entities. We believe that American innovation and American public-private partnerships are really the way to compete with the state-owned entities, and we look forward to engagement with all of our government stakeholders.

I'm going to now hand it off to Tara Neider, again, our Senior Vice President and project director for Natrium.

CHAIR REMPE: So I don't know if this is your first time at ACRS, but we like to ask questions.

MR. LEVESQUE: Please.

CHAIR REMPE: Are we allowed to ask you questions? I know you've got a meeting you've got to go --

MR. LEVESQUE: Absolutely. No, we look forward to that. Yes.

CHAIR REMPE: I'm just curious. I mean, you've mentioned that you've got five topical reports in the queue and you mentioned that you look forward to future engagements with ACRS. Do you have a time frame when you think you'll be back before ACRS?

MR. LEVESQUE: Well, we'll share, in further reports, we'll share our plans for our construction permit application, which will be early next year. We would plan multiple meetings. We certainly don't think it's too late or too early to be engaging you because we have a fully mobilized project. I know there's many projects that you can read about everyday in the U.S., but I would hazard to say maybe none of them have 800 design engineers working on a design, soil borings at the site already complete. I mean, this project is quite advanced and, given the construction permit application early next year, we think this is a great time for the first ACRS meeting.

George, Ryan, any --

MR. WILSON: This is George Wilson. I'm the VP of Regulatory Affairs for TerraPower. Right now, Billy Jessup is in the room, we have scheduled meetings --

MR. BLEY: Can you talk more into the microphone?

CHAIR REMPE: You need to be closer to the mike for the people online. I'm sorry.

MR. WILSON: This is George Wilson, TerraPower, VP of Regulatory Affairs. I know that we have some topical report meetings lined up with the ACRS starting in August, this August. But we wanted to come in, give an overview now, so that you guys could potentially ask questions and then we can, you know, as we go from there.

CHAIR REMPE: I'd caution you that it's not showing up, we have our schedule that we went through earlier today, and it's not on the schedule. So please work with our staff to make sure that it is as time that members know in advance, and they can make their plans accordingly if you're going to be coming in in August. That's why I asked --

MR. WILSON: That was the feedback that we had gotten from NRR. I think Branch Chief Billy Jessup is here, if he wants to add some more.

CHAIR REMPE: You can just stand and talk. We've gone with new technology. And be sure and say your name.

MR. JESSUP: Good afternoon. This is Bill Jessup, Chief of Advanced Reactor Licensing Branch 1, NRR. Our project managers are working very closely with ACRS staff as we progress through the review of the various topical reports and white papers, as well.

I know the ACRS is not looking at white papers. We are working very closely with ACRS staff to schedule forthcoming meetings that have been --

CHAIR REMPE: It's not on the August schedule. If we're going to meet in August, be sure and get it on the schedule, but that helps to know. And at some point, I know some of this proprietary, but some of your, if you can tell us which report it is that's going to be coming in in August, that would be helpful for us, too, to know. I don't know if you want to say it on the record here.

MR. WILSON: I think the NRC staff, we have a -- this is George Wilson again. We have a topical report. Our reactor design is a little bit different. We have a topical report on the nuclear island, energy island interfaces. Since we can actually operate the turbine, it's not dependent, we can operate the turbine at, like, 100-percent power and the reactor at 10-percent power. They're not linked like a light water reactor, so, based on that, there are certain regulations that will not apply to the Natrium design. So that is a topical report right now the NRC staff has completed the audit on and they're finishing up their review, and I know that will be one that's coming to ACRS.

We submitted principal design criteria. That will be another one that the NRC has just started reviewing. So those are ones that will be coming up to the ACRS first.

CHAIR REMPE: Okay. I did look over your regulatory engagement plan, but I just couldn't get a clue of what's coming in and when. So thank you.

MS. NEIDER: As was mentioned, I'm Tara Neider. I'm the project --

MEMBER PETTI: Microphone, please.

MS. NEIDER: Oh, okay. As Chris mentioned, I'm Tara Neider. I'm the project director for Natrium, and I wanted to talk a little bit about the Kemmerer site that we're selecting to build this plant on and then discuss a little bit about the team that we've put together, which I think is a pretty world class team.

As Chris mentioned, we are building the Natrium reactor at the site of a coal plant that is scheduled to shut down. And I can tell you that when we did our tour of various places to put the plant in Wyoming, this town, actually there's two towns, Diamondville and Kemmerer, they really welcomed us with open arms. It's really important for this community to have something to replace the work that was done at both the coal mine, you know, which feeds the coal plant, and then the coal plant. So it's been a very, very cooperative agreement we've had with the town. However, it is a small town, so we do have to make sure that we can be prepared for the number of people that we have which will be coming during the construction to build the plant.

The plant is a sodium fast reactor, and it's a pool type reactor, and most of the safety-related systems are underground. Mark and Eric will be talking later about the details of that design. But what really puts us apart from others is that the power generated from Natrium is dispatchable. Between the nuclear island and the energy island, nuclear island can run all the time, the energy island can actually go up and down in power as the power is needed on the grid.

Our team is, as I said, a world class team. Our design partner is GE-Hitachi. They have very strong experience in sodium fast reactors and, obviously, commercial BWR plants. And we also have Bechtel. Bechtel is doing the construction, and they're also responsible for the design of the energy island and some of the civil works on the nuclear island.

We also have a lot of other entities that are part of our team. Specifically, we've engaged with a number of nuclear utilities to provide us guidance on the operations piece and make sure that both operations and maintenance is considered in our design. Those include Duke Energy and Energy Northwest, as well as Pacific Corp, who will be the owner/operator of the plant eventually. Pacific Corp or, actually, a subsidiary of Rocky Mountain Power is the utility in that area, and they will eventually own and operate the plant, so they are part of our team moving forward.

We also have a number of national labs that are doing various support with regard to modes and methods and also testing of our advanced fuel for this reactor. And we have a number of other entities, as well, a number of universities. The University of Wisconsin, Oregon, and also NC State is part of our team. So we really feel that we've put together a pretty top-notch team to be able to move forward.

In addition to our teammates, we are trying to make sure that our plant meets the needs of the commercial market. And as a result of that, we've formed what we call the Natrium utility advisory group, but we also now call it users because there's a number of companies in the advisory committee that are not utilities at all. They're industry participants. And so those people we meet with on a regular basis, and they provide us guidance and take a look at how we're going through things. And I think we're up to about 30 people, 30 companies on that advisory group, so it's very positive.

We just joined INPO as just a supplier at this point, but we think that having joined INPO will actually provide us a lot of experience, especially the lessons learned from the industry going forward. And then, finally, we also have a memorandum of understanding with JAEA, the Japanese Atomic Energy Agency, and Mitsubishi. The reason that we have that memorandum in place is that they will be providing their own sodium fast reactors, but they also have a fair amount of experience running Joyo and Monju in Japan, so we really felt that their guidance would help us with this plant. And where they've helped us if they've participated in some of our design reviews going forward.

So those are my main points I wanted to make today, and I think, at this point, we'll ask if you have any questions or comments before we get into the technical side.

MR. BLEY: This is Dennis Bley. You've said you joined INPO. Are you considering being involved with FLEX and SAFER, or does that just not match up with you at all?

MS. NEIDER: We are considering FLEX and SAFER, yes. Not SAFER. FLEX, I know, we are considering a FLEX building, but it's going to depend on where our analysis comes up with things.

MR. BLEY: Thank you.

VICE CHAIR KIRCHNER: Maybe we'll get into this as we go forward, but, just at a very high level, just the first order, is this an evolution of a PRISM design that GE had been advocating back in the mid - late 80s time frame?

MS. NEIDER: There are certain elements of the PRISM design. We kind of combined the TerraPower traveling wave reactor and the GE PRISM reactor to develop the Natrium reactor. The reason, you know, the main change from those two is the size, but we did find some things on the various components that we did have to do differently just because --

VICE CHAIR KIRCHNER: When we get into closed session, maybe just keep this in mind. I'd like to ask some questions about what you learned from the PRISM design. In particular, the NRC back in that time frame, actually, a CER on the PRISM reactor, so it did get a fair amount of attention --

MS. NEIDER: Yes. And that safety evaluation report has been combed over by our engineers, and there were certain open items, like transient testing, that was required, and we've added that to our program. So we do believe those things are covered.

VICE CHAIR KIRCHNER: Good. Is this going to be the end of your open presentation, or are you going to shift gears here?

MR. WILLIAMS: We haven't actually started our open presentation.

(Laughter.)

VICE CHAIR KIRCHNER: Well, it's not unusual at ACRS to not get beyond the first --

(Laughter.)

VICE CHAIR KIRCHNER: -- so I'm just checking. Okay. Go ahead.

MR. WILLIAMS: Okay. I'll introduce myself real quick. My name is Eric Williams. I'm a Senior Vice President at TerraPower, and I'm the design authority for the Natrium project. And I'm going to be talking today about a plant design overview with three main points really to provide an overview of the plant design and operation, including both the nuclear island and the energy island. We're going to be focusing on more of the innovative features that we have, you know, comparing us to a light water reactor or something operating today, what would be the key differences for us, and also some differences between historical sodium fast reactors and the Natrium reactor, as well. And then I'll hand it off to Ryan, and he'll talk about the licensing strategy.

So this first slide to begin with, I'm going to spend a little bit of time on this one. We've talked about this one in all of our NRC engagements because it really has the key safety features all contained on one slide.

And so when you look at the Natrium reactor, it is a 840 megawatt thermal core output, and the key innovative features are really in that very first bullet. It's a pool type reactor, as opposed to a loop style reactor, so you have a large reactor vessel with a large liquid inventory of sodium within it. There are free surfaces within this reactor vessel, so it's a little different way of thinking from a loop reactor. It also uses metal fuel, a type of fuel that was undergoing heavy amounts of research and development in the U.S. legacy SFR program but never got turned into a full metal core until now. So there's various reactors out there that have used oxide fuel. I think even PRISM was intended to, you know, start out with some oxide fuel and convert to metal fuel. Natrium is going to start out with all metal fuel based on the research and development done at EBR-II and FFTF on this metal fuel type.

And then it's got a molten salt energy island, so that's definitely a new piece that enables us to store thermal energy in the energy island coming off the nuclear island and use that to generate electricity at a rate demanded by the grid, as Tara mentioned in her introduction. And it also provides another source of large liquid inventory that also adds, along with the liquid in the reactor vessel, sort of a buffer to any, you know, immediate challenges to core heat removal.

So due to those things, we'll get into that a little bit more as we go through these slides, and you'll see the effects of those major features. But those are the three big ones.

And a couple of other things about the metal fuel to keep in mind, it has a high degree of compatibility since it's, you know, metal fuel and metal coolant. And so through a lot of the tests, the amazing tests really that were done in the EBR-II reactor, they demonstrated that compatibility with the coolant, and so that benefits us a great deal in the safety area. And then it also behaves differently, you know, with regard to reactivity, feedback effects, and things like that.

We've also done something, we've removed a big source of sodium-water interaction. So a lot of historical SFRs have delivered heat from the sodium system directly into a steam generator to go convert that to steam and make electricity. And one of the major challenges to safety with that was this energetic reaction between liquid sodium and water. And so by imposing the energy island with a molten salt system, we've taken away that safety challenge between sodium and water, so that was a big improvement made just in the architecture of Natrium.

And I've also already referred to the large thermal inertia that kind of gives us that simplified response to abnormal events.

So if we look over to the right --

MR. BLEY: Dennis Bley. I think you mentioned it earlier, but the salt and the sodium are pretty compatible? They don't have any problems when they interact?

MR. WILLIAMS: They do, they do. I was going to talk about that a little bit when we get to the salt chemistry slide, but I'll go ahead and --

MR. BLEY: That's fine, that's fine. You can wait.

MR. WILLIAMS: All right. I'll mention it then. So if we look at the right side of this slide, we've just simplified this to the three fundamental safety functions of control, cool, and contain, and I'll kind of point out the major design features under each one of these. You can kind of see they follow a pattern of having a normal means of performing a safety function, a passive means of the safety function, plus an inherent means of providing the safety function.

And so for control, normal control, I think, is quite similar to what other reactors do. We have motor-driven control rods that can control reactivity and control power level during normal operation and during maneuvers. We also have a gravity driven control rod scram similar to other reactors. But when it gets to inherent reactivity control, it gets different because of the design of the metal fuel and the core restraint system, how the core is mechanically designed, there is a large amount of inherent negative reactivity feedback fundamentally in the core design, and so what that means is that we can look at these events that we call unprotected transients where, essentially, we postulate that none of the control rods go in by gravity and, in a certain number of anticipated operational occurrence type events, with none of those control rods going in, we can bring the reactor not down to full shutdown but it can sort of self-correct itself down to a low power level and stabilize just using the inherent reactivity feedback. So that is an effect that was demonstrated a lot in EBR-II transients and it's a factor in this core design, as well.

MR. BLEY: In terms of, you mentioned about 800 thermal --

MR. WILLIAMS: 840.

MR. BLEY: -- megawatts. Yes. So from a size standpoint, I am assuming that your team has optimized, this is a lot larger core and power than EBR-II, so you've optimized, I think Levesque is going to explore this in the closed session a little more, it's about as large as you can go, let me say it this way, and still get the leakage you need so that you get that reactivity feedback that you're looking for.

MR. WILLIAMS: I would say, we can get into more specifics in the closed session, but I would say it's not as large as you can go, but it's as large as you can go with current technology without having to take some leaps forward in other areas of the technology, which we do plan to do. So that's a good point.

MEMBER MARCH-LEUBA: So what's the enrichment on cycle length?

MR. WILLIAMS: Enrichment is HALEU, so we're up to, you know, 19 percent, just under 20.

MEMBER MARCH-LEUBA: And the cycle length that you expect?

MR. WILLIAMS: So the initial core loads are going to be one-year cycle lengths, and the reason for that is because our initial fuel type that we're starting the reactor with is something that we want to advance using the lead test assembly program. And we want to collect data, so we want to be taking fuel normally at normal intervals out of the reactor and sending it out for testing to form the qualification.

MEMBER MARCH-LEUBA: So testing on the full plan power.

MR. WILLIAMS: Exactly. So it will take several cycles of testing with the start-up fuel form to then collect enough data to qualify and go to the advanced fuel. And then cycles will expand to two years and a lot less number of assemblies needing four reloads.

MEMBER MARCH-LEUBA: And what's your break-in temperature?

MR. WILLIAMS: It's a 510-degree C outlet and 360-degree C inlet.

MEMBER PETTI: So the startup fuel is uranium-based, but the advanced fuel is the uranium plutonium?

MR. WILLIAMS: No, it's all uranium-based.

MEMBER PETTI: It's all uranium-based.

MR. WILLIAMS: Yes, all uranium-based. Yes, yes. In the closed session, I'll talk about the differences there. Any other questions on control?

Okay. So when we look at cooling, again, well, the main thing is to hold on to the coolant, so we don't have any penetrations through the reactor vessel. The only penetrations into the reactor vessel go through the reactor vessel head, so there's really, you know, low probability of causing a loss of coolant accident. I'll also talk a little bit on another slide about how the pressures of the systems are designed to force leaks inward towards the reactor vessel, another feature to prevent losses of inventory.

And then we have, you know, in addition to the primary heat transport system which, you know, delivers heat throughout the reactor vessel, it's an integrated primary system, so all of the components of the primary system are inside this reactor vessel in the most compact way that we can lay them out. We also have an intermediate system to this that's also a sodium system, so most historical SFRs do have an intermediate sodium system. We do, as well. And this system can remove heat through both, can remove heat forced circulation or through natural circulation. And it has within it a sodium-to-air heat exchanger, so it's a very effective way of removing heat when we're down in the refueling temperatures or modes like that where we can remove heat and totally decouple ourselves from the energy island entirely by isolating ourselves, in fact, and remove heat during refueling through this sodium-to-air heat exchanger.

And then in emergency situations where power isn't available --

MR. BLEY: I'm sorry. This is Dennis Bley again. Where is that thing located? Is it inside this containment picture you got here?

MR. WILLIAMS: Yes. Let me get to the next slide because I'll show you real clearly where that heat exchanger is located, yes. Just trying to introduce the functions here and then we'll see where the equipment is actually physically laid out as we go through the rest of the presentation.

And so, finally, that's a non-safety way of removing heat. When we go down to the safety means of decay heat removal during a design basis accident, we have something called the reactor air cooling system. This is a system that goes back to PRISM design, traces its lineage back there, and its way of removing heat through the outside wall of the vessel using natural draft air cooling. And so it's a system that's pretty robust because it's always on, so we're allowing a certain amount of heat to be lost through the system during normal operation to get the benefit of having this system always on and ready to remove heat in an accident scenario. And, in fact, because it's based on radiation heat transfer and, you know, as a function of temperature to the fourth power, it really doesn't start to kick in until you get a large enough temperature difference going. And so we're able to design this system to have pretty low parasitic heat losses during normal operation and then really crank in when a design basis accident happens. And so that's our really robust way of removing heat during design basis accidents.

And then, finally, on containment, one of the key features here is having low-pressure systems throughout the plant. And so there's really a low driving force for leakage going out the reactor vessel. And, of course, design basis accidents have an intact primary system, so we're not dealing with LOCAs and things like that, so it's really a matter of keeping leakage rates as low as we possibly can through the reactor vessel head.

One of the key features here with sodium is it's affinity for radionuclides. So the fission products release through failed fuel will have to flow through a large depth of sub-cooled sodium, which will remove the iodine and cesium and other things that will add to our mechanistic source term analysis. And then we have multiple radionuclide retention barriers, so you've got the ability of the fuel matrix itself to hold on to fission products, we've got the intact primary system, we've got the cover gas above the liquid sodium, and we've got several barriers outside of that, physical barriers to prevent, reduce releases, as well.

MEMBER PETTI: So you didn't specifically use the words functional containment as the SECY.

MR. WILLIAMS: Yes.

MEMBER PETTI: Do you think of it as a functional containment?

MR. WILLIAMS: Yes, yes.

MEMBER PETTI: Okay. And there's no additional containment building.

MR. WILLIAMS: Right, right.

MEMBER PETTI: Okay.

MR. WILLIAMS: It's formed by multiple structure systems and components, yes.

Okay. So if you put all of that together, that's where you get to a pretty simplified response to abnormal events. They really all follow the same pattern. There is a scram set point that gets triggers and a reliable reactor shutdown takes place, and then there's a smooth transition to natural circulation cooling. The reactor air cooling system kicks in. It takes a long time, actually, to heat up all of this liquid and start to kick in the reactor air cooling system, and that starts to remove decay heat, essentially, indefinitely because it's just using air. There's no need to replenish the ultimate heat sink or anything like that. It's a low-pressure functional containment that will continue to prevent uncontrolled releases of radionuclides, and there's really no reliance on the energy island for any safety functions. There's actually no reactor protection system parameters on the energy island, so there's nothing out on that side of the plant beyond the nuclear island that could threaten core cooling, so no scrams needed from that standpoint.

MEMBER MARCH-LEUBA: What is the red stuff, what's the green stuff, what's the blue stuff?

MR. WILLIAMS: Oh, yes, I do have another slide, I keep saying this, that will talk about that.

MEMBER MARCH-LEUBA: I don't believe you.

(Laughter.)

MR. WILLIAMS: You don't believe me? Okay.

MEMBER MARCH-LEUBA: But you keep talking about this air cooling, and I'm thinking it's inside of those green things.

MR. WILLIAMS: Okay. Yes, let me point out some of this now. The next slide I have talks about the flow path within the reactor vessel, so it will just go a little bit more detail. Okay. So we have, the reactor vessel is laid out, and it actually has multiple --

MEMBER MARCH-LEUBA: The only way to do it is with a mouse, yes.

MR. WILLIAMS: Yes, yes, look at that. All right. Okay. So there's actually physical barriers that separate the reactor vessel into two main pools that we call the hot pool or the cold pool. The red region is meant to indicate the hot pool, so that's what you see here from the top surface of the reactor vessel, it goes all the way down through this central region that we call the upper internal spectrum because it --

MEMBER MARCH-LEUBA: Is that like a core barrel in the --

MR. WILLIAMS: Yes, it's within the core and the core barrel and it's bounded by, even down here, by the core grid plate at the bottom of the core. And then that separates you from the blue region, which we call the cold pool, which goes from the intermediate heat exchanger exit all the way down to the reactor vessel. It's where the pumps take suction from, so the pumps will take suction from the cold pool and deliver it to the bottom of the core. We call that the high-pressure plenum at the bottom of the core, and then flow goes out, that little darker shaded red box is really the heated length region of the core. You can see how small that is relative to the overall height of sodium above it.

MEMBER MARCH-LEUBA: Give me a reference. How tall is that? Two meters, five meters?

MR. WILLIAMS: It's about 1.3 meters, and the overall height of the fuel assemblies are, like, 15 feet. I'm mixing units there. So flow comes up through the core and into this hot pool, and then it enters these heat exchangers. There's two of them. They're kidney-shaped heat exchangers you see on here, and the hot pool fluid is taken into these heat exchangers and flows down and discharges into the cold pool that's in blue. The pumps, like I said, take suction from there and return the flow to the core inlet.

So there's also free surfaces in the reactor vessel, the pool reactor. So there's a liner region that goes along the outside of the reactor vessel that is connected to the cold pool, so that means the reactor vessel wall is seeing cold pool temperatures and not hot pool temperatures, and it means that there's a level difference there that is equal to the unrecoverable pressure drop through the IHX. So there's a level difference there, and when you shut the pumps off that equalizes, yes.

So those are kind of some of the unique things about the way the reactor vessel is laid out. And in some of our analyses, we also call something a warm pool, but I won't get into that. That's just sort of an intermediate region that goes around the IHX.

VICE CHAIR KIRCHNER: The intermediate coolant loop is a salt loop, right?

MR. WILLIAMS: No, sodium.

VICE CHAIR KIRCHNER: So you've made the statement that the nuclear island's --

MR. BLEY: Use your mike, please, Walt.

VICE CHAIR KIRCHNER: Sorry. It's on Dennis. I'll get a little closer to it. That the nuclear island is separated from -- what do you call the balance of the plant here?

MR. WILLIAMS: Energy island.

VICE CHAIR KIRCHNER: Energy island.

MR. WILLIAMS: Yes.

VICE CHAIR KIRCHNER: Do you have to worry about freezing your sodium, or is the salt melting point in the energy island at a high enough temperature that you don't have to worry about overcooling transients? That's a way that that non-nuclear island could couple in an adverse manner with your nuclear island.

(Whereupon, the above-entitled matter went off the record at 1:38 p.m. and then went back on the record at 1:38 p.m.)

MR. WILLIAMS: Okay. All right. So there was, I think there was kind of two questions in there. One had to do with are you worried about freezing, and the other one about overcooling. And I think we are, of course, looking at overcooling as part of the safety analysis. You always have to do that, so, you know, pump overspeeds and heat exchanger being more effective than you intend them to be or turning on, blowers turning on when you don't expect them to are all accident initiators that we've looked at and don't see anything real problematic there, again, because of all that thermal inertia that we have.

Freezing is certainly something that we have to manage and engineer our way through. During normal operation, there's not a challenge to freezing because everything is nice and hot. But if you were shut down for an extended period of time, we have heat tracing on all of the sodium and salt piping to prevent freezing from happening, and it's also there to allow for maintenance because, when you want to do maintenance, you do want to freeze to actually allow you to perform maintenance. So there's a lot of heat tracing in the reactor for that reason. It's pretty simple but --

MEMBER HALNON: The energy island, though, if it freezes up, it's just an economic issue, it's not a safety issue, correct?

MR. WILLIAMS: Right. It's not because we would run back and decouple, close the isolation valves entirely.

MEMBER HALNON: Yes, the inherent self island, self nuclear island.

MR. WILLIAMS: Right. Okay. So there was a question on ancestries --

MR. BLEY: This is Dennis Bley. I don't know if you guys know it, but the meeting timed out about five minutes ago and we can only listen on the phone now. We can't --

CHAIR REMPE: Dennis, Dennis, most people have rejoined. There was something that just happened with the software and we got all kicked off for something, but, anyway, most people have been able to rejoin. And I think the court reporter, the court reporter, just to verify, you can hear, right? Right. Thank you.

MR. WILLIAMS: Okay. Yes, so if you look

back at all the sodium fast reactors that form a basis for Natrium, you can go all the way back to EBR-I, although that was really just a demonstration of breeding and wasn't really sodium cooled, it was sodium potassium cooled. But EBR-II and FFTF are the key ones for us. EBR-II with a lot of the passive and inherent safety tests, the run beyond cladding breach tests. And FFTF certainly in terms of all the operating experience with sodium systems and also the tests that they were performing on metal fuel and scram to natural circulation tests. And then even TREAT, the transient reactor test facility, is a source of information on severe accidents with metal fuel.

So those are the main areas to look at for the ancestry for Natrium. And like was said earlier, it's a combination of the PRISM and the traveling wave reactors, and I also have to mention Clinch River because the Clinch River breeder reactor had certain parts of its design that are beneficial to us, as well.

All right. We spent a lot of time on that slide, but there's a lot on there, so I think that was good. So I think I've talked about most of this, you know. Key differences from light water reactors, we've kind of already discussed that. The systems are very compact. You'll see that on the next few slides visually. The low-pressure systems, the efficient heat transfer just because you've got metal fuel, metal coolant, a metal reactor vessel. It's metal, metal, metal, so it's very efficient if you draw the thermal resistance diagram. Pool design with a large coolant inventory is a big factor in safety. Modularity, we've used modular construction wherever we can. It's not a factory-built reactor vessel, but we still use modularity in a lot of the design of the plant and it's architected to really separate safety and non-safety with special treatment systems away from non-safety systems.

And also that energy island, nuclear island separation allows for parallel construction techniques, and the extra inherent and passive safety allows us to do a reduced emergency planning zone.

MEMBER PETTI: So is it a goal for the EPZ to be the site boundary?

MR. WILLIAMS: Yes. All right. So this slide gives you a good view visually of the layout. This is pretty true to what this area in Wyoming actually looks like, so pretty nice view of the plant. And you can see in the center of the slide, I'll use my mouse here again, we have the reactor, the three main nuclear island buildings. There's the reactor building in the middle, there's the reactor auxiliary building just next to it, and the fuel handling building just next to that.

So those three buildings form the central point of the nuclear island, and the heat essentially comes from below grade in the reactor building out of the reactor vessel through those intermediate heat exchangers that we saw in the other slide, enters the intermediate heat transport system, which is mostly located in the reactor auxiliary building, so the intermediate sodium goes through these sodium to salt heat exchangers and delivers the heat to the salt piping, which leaves the reactor auxiliary building where my mouse is going all the way over to the thermal energy storage tanks, which are over here on the energy island. The thermal energy storage tanks, well, the hot tank essentially pumps salt into the salt-to-water steam generators that are in the steam generator building. Those will generate super heated steam to go to the turbine.

And everything that you see beyond the energy storage tanks are sized for 500 megawatts electric, even though the nuclear island is generating

345 megawatts electric steady state. So that allows us to ramp up to 500 megawatts electric for about five hours if the grid demands it, but it also allows us to go down to about 100 megawatts electric if the grid demands it and the renewables are working the way they're supposed to and the sun is shining and the wind is blowing, you may want to reduce that heat and save it up for later. So that's the magic of the energy island and the ability to load follow with that while keeping the nuclear island constant.

MEMBER HALNON: A couple of things jump out. First of all, you may have your reasons, but I would put the control building more centralized on the other side of the plant just so you can get to energy island quicker. But, nevertheless, if you had a two unit or three or four multiple unit site, what synergies would you have with this single unit besides the switch yard?

MR. WILLIAMS: Yes, exactly. So the dual unit site would essentially share the fuel handling building, and so, in fact, one of our requirements is to design the fuel handling building so that a later second unit could be added at a later time.

MEMBER HALNON: Just on the other side.

MR. WILLIAMS: Just on the other side.

Exactly, yes. And they can also share the energy island. Our site is Wyoming is actually sized to be able to add an additional pair of energy storage tanks.

So that's the other thing you can do is the energy island is very modular. There's a lot of optionality on the energy island, depending on the grid and where you're locating the plant, to have multiple salt tank pairs, have a different sized turbine, have multiple turbines. There's lots of things you can do with that to meet whatever that grid demands.

MEMBER HALNON: So just back to my first comment, is there a reason why the control building is not centralized? Just as an ex-operator, I would want to be able to get to the energy island rather quickly.

MR. WILLIAMS: I think that's a good comment. I mean, I know there were a lot of factors into placing that. I don't recall exactly. I mean, it was safety of providing the cabling that goes from the control building into the reactor building. It also houses the reactor protection system equipment in there, so I know there were a lot of factors in that, but I don't recall exactly why it's there.

MR. WERNER: I could maybe help a little

bit. This is Mark Werner from TerraPower. One thing to note is that the operators in our control room don't have any safety-related actions, so they don't need to be close to anything to do anything. We do have control, something like a control room in the energy island to provide local control of, you know, near the turbine facility, so there will be people in that area that will provide local control. We are a dispatchable power type of plant, so there's a little bit less to do over there.

But one of the things we did try to do on this plant, as Eric mentioned, is a little bit more of a distributed architecture. From a construction standpoint, that control room can be constructed with a work front at the same time as the other facilities. And we see a lot of value in shortening our overall construction duration from a delivery standpoint. So by separating all these buildings, providing a little bit of space, we can have four or five separate crews working on facilities at the same time. If you put them all on top of each other, then you have to wait for the mechanical team to finish, the civil team to finish, before you can start laying down your electrical work. By separating them, they can be working at the same time, shortening construction.

MEMBER HALNON: Thanks. Acreage-wise, what is that about, what we're looking at? Five, five acres?

MR. WERNER: The overall site is about 50 acres. The nuclear island itself is about 16 acres, and it's still fairly distributed, you know. Eric mentioned kind of the flexibility of our layout and flexibility of our energy island, you know. How we orient all of these facilities is really not coupled to our safety case at all, and so if we had to deploy a site not like Wyoming and space was a real concern, we could make things closer.

VICE CHAIR KIRCHNER: You want to orient your turbine building in a preferential way.

MR. WERNER: Sure. Right. And to be clear, there are spacing requirements and location requirements. I'm just noting that, you know, if we go to a site that was narrower or longer, because we're decoupled and that salt pipe doesn't lose a lot of heat, isn't that expensive in the grand scheme of things, we can reorient our two sites to kind of fit the local geography pretty well.

VICE CHAIR KIRCHNER: Dennis, I see your hand up. Go ahead.

MR. BLEY: Yes, I had a couple of things.

One is I kind of agree with Greg. I don't know how long the distance is from the control building over to the energy island, but having a lot of separation from the operators could be troublesome at some times. I understand your construction ideas, too.

The idea is, I really like this. This is pretty clever. The other side of the story, if the grid load drops off or the grid trips, it would appear you don't need to scram, you can keep pumping heat over into those energy storage tanks. How long can you do that before you got to ramp down power or trip the reactor?

MR. WILLIAMS: Yes, it's essentially another, it's just five hours in both directions.

MR. BLEY: Either way. Okay.

MR. WILLIAMS: Yes.

MR. BLEY: And you run into some kind of mechanical limits on the temperature if you --

MR. WILLIAMS: Right. We would be looking at the tank levels, and the tank levels would get to a minimum or maximum --

MR. BLEY: Sure.

MR. WILLIAMS: Yes.

MR. BLEY: Okay. Oh, by the way, please keep talking into your microphones. When you turn your head away, we can't hear you out here.

MR. WILLIAMS: Oh, okay. Thanks.

MEMBER MARCH-LEUBA: We're having problems today that we typically don't have. On the storage tanks, do you increase the temperature of the salt or do you melt more salt? You said something about level.

MR. WILLIAMS: Yes, we don't melt more salt. The salt tank levels change, so, if you are ramping down electrical output, you accumulate hot salt in the hot salt tank, so the level goes up.

MEMBER MARCH-LEUBA: So it's just expansion.

MR. WILLIAMS: Yes.

MEMBER MARCH-LEUBA: So you're changing the temperature of the salt, not the amount of salt.

MR. WILLIAMS: It's the amount of salt in that hot salt tank, yes. It will go up. You will accumulate hot salt if you're throttling back the hot salt --

MEMBER MARCH-LEUBA: So what is it the cold salt?

MR. WILLIAMS: And the cold salt is in the cold salt tank --

MEMBER MARCH-LEUBA: Oh, so you have two

tanks, one is cold, one is hot.

MR. WILLIAMS: Yes.

MEMBER MARCH-LEUBA: So you are, it's not that you're melting salt.

MR. WILLIAMS: Right.

MEMBER MARCH-LEUBA: You're transferring.

MR. WILLIAMS: Moving it, yes, yes. It's a constant volume of salt.

MR. LEVESQUE: It just, you know, it transfers from one tank to the other as we turn hot salt into cold salt.

MR. WILLIAMS: Dr. Rempe, what do we do when we get to 2:00? Do you want me to keep going?

VICE CHAIR KIRCHNER: We'll stop where we are, and then we will have people who are joining us and are okay to join us will come in on a different Teams connection --

MR. WILLIAMS: Okay.

VICE CHAIR KIRCHNER: -- that's been limited. So we'll need to take a break, and we'll need to just make sure everyone in the room, I see you've handed out already proprietary information. I hope you've got a handle on it because the room has been open, but we'll close the room, and then we'll go into the closed session.

So if you're at that juncture, this would probably be a good time for us to take a 10- to 15-minute break and reset our Teams connection.

MEMBER MARCH-LEUBA: It doesn't need to be exactly at 2:00. It can be at 2:30.

CHAIR REMPE: I hadn't heard it had to be exactly.

MR. MOORE: This is Scott Moore, the executive director. It does not have to be precisely at that time. The meeting in Teams will remain open.

VICE CHAIR KIRCHNER: Okay.

CHAIR REMPE: But when you want to go to the closed meeting, it doesn't happen quickly because we've got to check the room and a break is good idea.

MR. WILLIAMS: Okay. All right. So this was the vertical cut-through, those central three nuclear island buildings, and so right away you can kind of see that we have the, like I said, the safety related and non-safety with special treatment systems below grade. All the radiological systems are essentially below grade in the reactor. That includes the reactor vessel, it includes the piping of the intermediate heat transport system just above the reactor vessel. And this area that you see, this rectangle above the reactor vessel is what we call the head access area, so there's a lot of equipment, very important equipment in that region.

Over below grade in the reactor auxiliary building, we've got radiological systems there for processing both the liquid sodium from the reactor vessel and the gaseous, the gases from the reactor vessel, so those are all below grade. And then above grade, you've got the sodium-to-salt heat exchangers, essentially, and the intermediate sodium pumps that make up the intermediate heat transport system.

You can also see on the right the fuel handling building, spent fuel pool below grade. And then in both the reactor building and fuel handling buildings, you can kind of see they're very open structures, so the refueling maneuvers can happen between these two buildings. It's not that we've forgotten to add systems in these buildings. They really are open. And you can see that, because the radiological systems are below grade in this nuclear seismic grade concrete structures, we can afford to take a different approach above grade, and we have, essentially, steel structures above grade. So that's where a lot of the modularity comes in, too. These steel structures, you know, can be handled much differently and economically.

I had a question earlier about where was the intermediate air cooling heat exchanger, and this was the slide that shows that very clearly. Over here on the side of the reactor auxiliary building, outside of the reactor auxiliary building, and so that is a sodium-to-air heat exchanger connected to the intermediate heat transport loop and it allows through both forced cooling and natural draft cooling to take heat out of the reactor vessel and deliver it to the air without using the safety related system. And so that's the normal decay heat removal system that we would use in refueling modes and hot standby and so forth.

You can also see the ducts for the reactor air cooling system here, and so the air is flowing down and around the reactor vessel and it's coming in through ducts, and these lower ducts is where it's coming in and it's coming out through the taller ducts. And there's four inlets and four outlets to that system.

VICE CHAIR KIRCHNER: What parts of the reactor building do you have to harden against natural hazards in particular?

MR. WILLIAMS: Yes, that would be this level at grade that protects the --

VICE CHAIR KIRCHNER: So your reactor air-cooling ducts, if they were severed for whatever reason, the system would still function?

MR. WILLIAMS: Right. We are looking at that, and they're set up on different sides of the reactor building with enough separation that we can deal with some of those hazards.

VICE CHAIR KIRCHNER: One of the vulnerabilities that was, I think, there in the PRISM design was that they had that functional heat exchanger above grade, potentially exposed to hazards. So, basically, if these ducts sever, you'll still get a natural draft through the reactor cavity to take out the decay heat.

MR. WILLIAMS: They are very robust with regards to, you know, if you sever two of them and you still have two functioning or if you increase pressure drop in these ducts. A lot of the testing that was done on the system was done at Argonne National Lab and showed that it really wasn't as sensitive to pressure drop as you would think. It's more radiation heat transfer limited, and so it's a very unique kind of robust system. But there's all kinds of external hazards also that can degrade the system that we look at, you know, debris getting entrained into the ducts, and there's all sorts of degradation mechanisms that were taken into account to --

VICE CHAIR KIRCHNER: Dennis, do you have your hand up again?

MR. BLEY: I do. This is Dennis Bley.

VICE CHAIR KIRCHNER: Okay. Go ahead.

MR. BLEY: It looks like if you've hardened the aux building, things are probably pretty good, except airplane crash, I assume, could take out the intermediate air cooling system; is that right?

MR. WILLIAMS: The intermediate, there's two intermediate air cooling heat exchangers on sides of the reactor aux building, so I'm not, I can't answer that exactly whether it can take out both of them or not; I'd have to go back and look. But those aren't safety-related systems either.

MEMBER MARCH-LEUBA: Your safety-related air cooling is inside of the reactor cavity.

MR. WILLIAMS: Right.

MEMBER MARCH-LEUBA: And where does the heat go?

MR. WILLIAMS: So the heat --

MR. BLEY: Yes, you were going to show us that.

MR. WILLIAMS: Yes. So the heat, it comes in through the reactor air-cooling ducts that I'm pointing to with my mouse. There's two of them shown here. There would be two, a mirror image, on the other side. So four air intake ducts.

MEMBER MARCH-LEUBA: That reactor cavity has a couple of holes that are connected with the atmosphere.

MR. WILLIAMS: Absolutely. It's all connected to the atmosphere. And so it comes down --

MEMBER MARCH-LEUBA: The one, it looks like a containment fluoride. It's not.

MR. WILLIAMS: Right. Well, yes. So let me try to --

MEMBER MARCH-LEUBA: You have a way for -- I couldn't figure out where the heat was going.

MR. WILLIAMS: Okay. We do have more sketches of this in the closed session that show the details.

MEMBER PETTI: Do you use a seismic isolator on the --

MR. WILLIAMS: We don't seismically isolate the plant at large, but we have seismic isolation on the reactor vessel.

MEMBER PETTI: The reactor vessel. Good.

MR. BLEY: I guess, before you leave this, I'm still thinking about an aircraft crash, and that reactor building up there, that's a steel building it looks like. If that gets taken out, it seems like those air ducts and things could all get crushed up and wouldn't work anymore.

MR. WILLIAMS: Yes, we'll have to show that there's enough air cooling even with an airplane crash.

VICE CHAIR KIRCHNER: But you wouldn't want collateral damage if you have an aircraft event hitting that sodium intermediate loop and then spreading sodium into the reactor building cavity.

MEMBER MARCH-LEUBA: On a rainy day.

MR. WILLIAMS: Yes, on a rainy day.

MR. BLEY: Okay. We'll see that kind of detail later, but I'm curious about how you deal with that.

MR. WILLIAMS: Oh, Tara did mention something to me real quick. There's been an evolution to this design, and the reactor, it looks like the reactor air-cooling ducts are integral with the reactor building, but they're actually separate from the building. So the reactor air-cooling ducts are safety-related ducts, and the reactor building is not. So those are separate.

MR. BLEY: Okay. There's a great big crane up there that could come down with the building, though, it looks like. Pretty heavy.

MR. WILLIAMS: We'll jump into 3D model in the closed session, and we can look at this in detail of the current design. This is a pre-conceptual image.

Okay. So I'll go through this one a little quickly. We talked a bit about the energy island thermal storage. The great part about this side of the plant is that all of this technology is readily available from the concentrated solar plant industry, and so that was one of the benefits of why we chose it. We even chose the same salt composition that is typically used in concentrated solar plants to make their, you know, good access to the OE database of all of that work.

So when you look at this, there's lots of different options, from the number of tanks, the number of steam generator trains, and what sized turbine you need for the grid where you want to put this plant. The picture on this slide on the left kind of shows you a top-down view of a concentrated solar plant, and you can see how many salt tank pairs they have. So they've definitely chosen to go with a higher degree of thermal energy storage for this location, and, like we said earlier, that's something that you can do with the Natrium design, as well.

All right. So a few things to just talk about the benefits of sodium and then the benefits of salt. When we look at sodium coolant, and there's been a lot of work with liquid sodium and liquid sodium reactors in the United States, the one that's really not, that's not really mentioned on this slide is that sodium is chosen for its neutronic properties because it doesn't moderate neutrons. So if it were to have a higher degree of moderation, it would soften the spectrum, and we may not have a fast reactor anymore. So it's chosen for the reasons of wanting the sodium fast reactor, and then the higher heat capacity that it provides allows us to have reasonable sized pumps. The high heat transfer is really the most important thing. It allows the core to be reasonable sized, and it allows that great decay heat removal that I talked about and allows us to use these sodium-to-air heat exchangers that are so effective and so valuable to us in decay heat removal.

It's also got a good range for where we want to operate, so the high boiling point around 883 degrees C and a melting point around 98 degrees C, and all of that is at atmospheric pressure, so that means we can have a high temperature heat output coming from the reactor vessel, which allows us to have high thermal efficiency without having to pressurize it. So that's the key to being at atmospheric pressure.

And then the density of sodium is remarkably similar to water, which allows us to do a lot of testing, do the scaling of a test for sodium in a water system and actually get that to work, and so reducing the costing of testing and increasing the value of testing with water. Lack of corrosion, so there's been demonstrations of very low corrosion in sodium systems over time. We learned a lot of that from FFTF. Limited auxiliaries, like I mentioned, a couple of systems to remove non-condensable gases and impurities from the sodium liquid and the gas; but other than that, there's not a whole lot of conditioning that you have to do.

And then there's a large sodium inventory, so 800 cubic meters of sodium in the reactor to absorb all of that decay heat removal when you need it. And the graphs here show you the differences between, like, the peak, the cladding temperature and the peak central temperature of the fuel between an oxide fuel and metallic fuel. And so because metallic fuel can remove heat so effectively from the fuel itself, you actually have a pretty close delta between the fuel temperature and the cladding temperature shown on the right, whereas an oxide fuel you don't. You have a higher center line fuel temperature that is sometimes above the boiling temperature of the coolant. And in sodium reactor oxide fuel reactors, they have that. So the normal operating fuel temperature was higher than the boiling temperature, but in Natrium it's several hundred degrees below the boiling temperature because of metal fuel. So those were some of the great benefits of using sodium.

And then this next slide just shows some pictures. If you come out and visit us in our lab, we'll let you take a look at these up close but not too close, and we even let people cut into solid sodium to kind of get a sense of what it feels like. It kind of feels like cutting through butter taken out of the refrigerator or something. But you can see it's opaque, so that's kind of an important property that we have to be concerned with in terms of refueling. A lot of times, it builds up an oxide layer on the top. It has an auto-ignition temperature that varies quite widely between about 100 degrees Celsius and 400 degrees Celsius. So, you know, you could have sodium sitting there below its auto-ignition temperature, but, of course, at reactor temperatures, it's going to be above that.

And on the solid side, you can kind of see it's built up this oxide film. It's kind of grayish or even pinkish sometimes.

And so one of the things with sodium is it does react energetically with both water and air, and so, for our sodium systems, we have to prevent and be prepared to detect and mitigate any sodium fires. So one of the downsides to using sodium is, in addition to the heat tracing we mentioned earlier, is providing all that sodium fire protection.

And so, you know, it's a kind of engineered safety system that has to exist in the plant. But the good news is it's all at atmospheric pressure, so, when we have leaks, they're not, you know, you sometimes see in the literature spray fires and pool fires and things like that. What we typically call it is a drift fire. It would be like a slow leak at a flange or something like that. Typically, these leaks are, you know, they don't get through the guard piping and inerted space that we have the pipe in, and so they're detected by the sodium leak detectors, which then allow us to go in there and perform the maintenance --

MEMBER PETTI: So all the pipes are guarded.

MR. WILLIAMS: All the pipes have some kind of, and we have a slide on this, some kind of leak protection to them. There's all different ways depending on where the pipe is.

MEMBER PETTI: The opacity I was aware of, but the program inspection and, you know, how you see into this code requirements. I know, obviously, some of the components are above the level, but you probably still have a need to get down lower with the sodium. Has the technology evolved that you've got a solution?

MR. WILLIAMS: We're still looking at, you know, technologies for under sodium viewing, so that is certainly a part of a project is to look into those. But we're also looking at, you know, risk-informed inspections and different ways of inspecting equipment that is below the level of sodium, so it's kind of a combination of all of them. But for the refueling, you know, we will have discrimination features on the top of the fuel assemblies that will allow all of that to occur, even under the opaque sodium. And fuel assemblies are inspected when they come out and they get cleaned and all of that, as well.

MEMBER BALLINGER: This is Ron Ballinger. Are you in any way connected with or taking advantage of the French side, especially with respect to inspection, refueling, those kinds of things with Phoenix and Super Phoenix?

MR. WILLIAMS: We're still working on that front. We'd like to be more connected. I don't think we are -- but we are talking right now.

MR. WERNER: I think we're sending a team over, I think we're sending a team over, like, next month to collaborate on methods and operations, yes. And this is Mark Werner.

VICE CHAIR KIRCHNER: Also look at construction, too, because if you study the Super Phoenix experience.

MEMBER BALLINGER: They had some infamous issues, like not being able to find the fuel elements.

MR. WILLIAMS: Okay. That would be infamous. All right. So another couple of slides just on salt. So we are using nitrate salt, you know, 60-percent sodium nitrate, 40-percent potassium nitrate. This is the typical composition used in the solar industry. Molten salt inventory is quite large with those large thermal storage tanks, 30,000 tons. So there's quite a bit of thermal energy storage capacity that we have over there, and, of course, anything happening on the other side of those tanks with the steam generators, the turbine, or all of that equipment, just can't make its way back to the nuclear island very quickly.

So there's lots of benefits of using this molten salt. It actually does have pretty good long life with really low performance degradation, and it's really good in the range of temperatures that we want. We can use carbon steel up until about 400 degrees C and stainless steel up to about 600 degrees C, so the cold tank side is typically carbon steel and the hot salt tank side is typically stainless steel. So a really good temperature range for us and a really high degree of efficiency in terms of retaining the heat. And so a lot of people ask us, you know, are you losing a lot of heat from the tanks that are sitting there, you know, during outages and things like that and not being used, and the answer is no. Very little heat is lost through them.

And so, you know, a really common use in the industry. I don't think it's been used in a nuclear facility, so, you know, we are taking a look at, you know, the operating experience and history with designing these tanks in the industry and making sure that they are designed to the right level of reliability for us.

MEMBER BALLINGER: Do you have any flammability issues with this salt mixture?

MR. WILLIAMS: No. So salt is not flammable with air, so when salt leaks, you know, you go and clean it up with a shovel. It solidifies on the outside, and there's no danger with that.

I was going to mention the salt and the sodium is an exothermic reaction. It's relatively mild. It's nothing like sodium and water, but it is one of the technology factors that we are doing quite a bit of testing on right now in our lab in Washington. And so we have a couple of systems, a couple of test loops running right now with sodium-salt interaction tests, and what the data are really showing us is that when you have, in our case you would have salt leaking into a sodium system, it does generate some energy, but it is relatively mild. And because the sodium is so thermally conductive, the heat dissipates very quickly in the sodium system, and you can detect it by the presence of, I think it's hydrogen and nitrogen that are generated from this reaction, and those are gasses that we're already looking at in the intermediate heat transport system. So you can detect a leak when it happens, and you can shut down and perform maintenance. You can train the heat exchanger and perform maintenance on that. But we're still continuing to test that.

Yes, go ahead.

VICE CHAIR KIRCHNER: Go ahead, Dennis.

MR. BLEY: You said hydrogen. This is Dennis Bley. I don't see any hydrogen.

MR. WILLIAMS: It's the -- oh, yes, you're right. Yes, it must have been nitrogen. Sorry. You're right. There's no hydrogen in there. Thank you for correcting me on that.

MEMBER PETTI: And there's no chemical issues in terms of the vapor pressure that workers need protection, besides the temperature of the salt, in terms of maintenance.

MR. WILLIAMS: Yes, I don't, I'm not aware of any.

MEMBER BALLINGER: There was an incident at Cadarache where they were cleaning a tank which formerly contained sodium and had residue in it, and something happened. Now, I'm a little fuzzy on what they were using, whether it was a cleaning fluid or whatever it was, reacted with the residual sodium, and they killed a few people.

MR. WILLIAMS: Okay.

MEMBER BALLINGER: It was an enclosed tank, and it was a bad hair day.

MR. WILLIAMS: Yes.

MEMBER BALLINGER: So there are those issues.

MR. WILLIAMS: Yes. I think for sodium you have to take a lot more precautions than with the salt.

MEMBER BALLINGER: Certainly with cleaning, with what they were doing.

MR. WILLIAMS: Right, right. Good comment.

So salt in the molten state, you know, it's the white substance on the left as a solid and a clear liquid as you can see on the right. And so, you know, leaks are, again, something that we have to consider and look at, but nothing like sodium leaks. So we'll definitely be looking at that and having plans for maintenance and recovery after leaks.

MR. WERNER: Hey, Eric, this is Mark Werner, just jumping in. If you go back a slide, just to maybe add a little more context to this. The image on the left is actually salt that we leaked out into a pan and just collected in a jar. And so when we're worried about leaks, it just solidifies to look like what's on the left. And on the right, you know, that's molten salt, but it's in an air environment. It's not in a glove box or anything, so this kind of image shows how compatible this working fluid is with the environment.

MR. WILLIAMS: Thanks, Mark. All right. So when you take a look at the safety systems, we called them, you know, compact and robust for the Natrium reactor. So if you just look at comparison to a light water reactor where you're, you know, your safety systems are daisy-chaining multiple systems to take the heat out of the reactor and take it to its ultimate heat sink, it just generates a lot of, we call it nuclear sprawl sometimes that makes a large fraction of the plant become safety related as a consequence. And so by having these reactor air-cooling ducts removing heat using air, you have a more compact system that doesn't result in that same sprawl. But having the integrated primary system all inside the reactor vessel, we've eliminated a lot of Section III pipe welds.

Being at atmospheric pressure we've

mentioned is a big feature here. And just that unlimited supply of the ultimate heat sink, very important for long-term cooling. Being fully passive, having a system actually always in operation, and just having a very rugged system. So there's a lot of good benefits to the technology in that area.

All right. So I'm getting close to the end here.

MR. BLEY: Dennis Bley again. I almost forgot my question when the picture went away. How hot does the sodium have to get -- I could have done the calculation, I guess, but I didn't -- before this transfer, heat transfer mechanism becomes real effective? It's got to go up a hell of a lot, I would think.

MR. WILLIAMS: Yes, yes, it's probably going to go up, you know, 50 to 100 degrees, I would guess. You know, it's always going to be ramping up as temperature is going up. It just takes a long time for that much inventory to raise its temperature. But the initial, you know, following an accident with only reactor air cooling available and nothing else, you know, you would see a lot of the decay heat turning into sensible heat, increasing the temperature of that liquid pool. And then, you know, over the course of a day or more, you would start to see it start to kick in more.

MR. BLEY: Okay. That gives me a hint anyway.

MR. WILLIAMS: Okay. So this was the one I was promising about the flow path. It shows kind of what I described earlier, but I think it gives you just a little bit of a clear image between the red hot pool and the blue cold pool. And the arrows show you the flow path through the heat exchangers and the pumps.

So a couple of features I'll just point to that we haven't talked about yet is there is a guard vessel outside of the reactor vessel. So one of the ways of preventing, you know, if there was a leak, and this would be a beyond design basis leak from the reactor vessel, we don't want to have interaction with air. So there's a guard vessel surrounding the reactor vessel and an inerted space in between them. And then the gap between that reactor vessel and guard vessel is sized such that, if there was a reactor vessel leak, the total liquid level would still remain above the pumps and the heat exchangers inside the reactor vessel. So even if you had a reactor vessel leak, you'd keep those components covered and you provide the opportunity to continue to use forced cooling, if you have it, and to still allow natural circulation cooling to occur, of course.

MEMBER MARCH-LEUBA: Are the pumps on the outlet of the heat exchanger?

MR. WILLIAMS: The pumps take suction from the cold pool and discharge to the core inlet.

MEMBER MARCH-LEUBA: Kind of push it. And what drives it through the heat exchanger?

MR. WILLIAMS: The heat exchanger takes input from the hot pool at the top and then discharges it into the cold pool.

MEMBER MARCH-LEUBA: By gravity?

MR. WILLIAMS: Through the pump. The pump is forced cooling up through the core --

MEMBER MARCH-LEUBA: So the pump is cooling on it.

MR. WILLIAMS: Right.

MR. BLEY: Hey, this is Dennis Bley again. On your first picture, the label on the guard vessel said containment. It looks like it is essentially a containment. Why don't you call it one?

MR. WILLIAMS: It's a major part of the functional containment strategy. The guard vessel surrounds completely the guard vessel, but it doesn't go up around the reactor vessel head. And so you have the seals in the reactor vessel head for the, you know, for the control rod drive mechanisms, and the other penetrations are also part of that functional containment.

MR. BLEY: Okay. This picture is not a real picture, so it looks like the vessel head is actually bolted down on to the guard vessel. Is that true, or is that just an artifact?

MR. WILLIAMS: Maybe Mark could help me with the interface there between the guard vessel and the reactor vessel.

MR. WERNER: Yes. This is Mark Werner from TerraPower. There will be some type of seal between the guard vessel and the reactor vessel. The reactor head will be bolted down on to the reactor support structure, but the guard vessel will be, you know, sealed to the reactor vessel to maintain that boundary.

MR. BLEY: Okay.

MR. WERNER: We need to keep that inert, as well, and we monitor that space.

MR. WILLIAMS: All right. You can see, the only other piece I wanted to point out here was the in-vessel transfer machine shown in blue there and pointed out with the label. So we are doing refueling inside the reactor vessel. Another kind of new thing, the reactor vessel head doesn't come off. New fuel comes in through an opening in the reactor vessel head and gets manipulated with the in-vessel transfer machine to its location; and, you know, spent fuel is manipulated with the in-vessel transfer machine to remove it from the core. We actually store fuel assemblies after they come out of the core for a period of time in in-vessel storage locations, which you can think of as a couple of rings of empty fuel assembly slots outside the core barrel. So they stay there, they cool off for a cycle, and then the in-vessel transfer machine removes them out of the reactor vessel. So that's a unique part of the technology, and so we chose to do it with an in-vessel transfer machine here for Natrium.

MR. BLEY: This is Dennis Bley again. In the closed session, are you going to be able to show us how that thing actually works? Does it index around or what's going on there?

MR. WILLIAMS: Yes. We've got a movie for you in the closed session to show that.

MR. BLEY: Great. I can't wait.

MEMBER BALLINGER: This is similar to

Phoenix.

MR. WILLIAMS: I think this is a pantograph, and I'm not sure if they use that at Phoenix. But FFTF used an offset arm type of equipment, and so it was different than this. And so we chose to go with this because we think it will allow us to have more, you know, meet the expectations of commercial refueling times, as opposed to what they did at a test reactor. All right.

VICE CHAIR KIRCHNER: So in the closed session, Eric, could you show us where actually in the core the spent fuel is stored temporarily?

MR. WILLIAMS: Yes, yes, we have that. So the last slide here before I turn it over to Ryan, we've kind of talked about all of this, but just to wrap up the overall story for heat removal is, you know, you have normal ways of removing heat, which is forced flow through the intermediate heat transport system, using that intermediate air heat exchanger. You also have a passive way of using that same system by not having forced flow and just using natural draft through the intermediate heat transport system, and that intermediate air-cooling heat exchanger has dampers that open and a blower that can also come on and force cool it if you're in the normal mode. And then if you're in the passive mode, the dampers just open and it allows natural draft cooling using the same heat exchanger.

And then, finally, I call it the inherent decay heat removal, and that's with the reactor air-cooling system because it's always on. And that's our safety-related decay heat removal method.

So that just kind of summarizes all the ways of getting heat out of the reactor, and now I will turn it over to Ryan who will talk about our licensing strategy.

MR. SPRENGEL: Good afternoon. Ryan Sprengel. I'm the Director of Licensing for Natrium. I did check before the meeting. It's been seven and a half years since I've been in front of the Committee, so I see some familiar faces and some new faces. Dare I say, I'm happy to be back in front of you.

It's been touched on already, but we are using a Part 50 licensing process. So we'll go through a two-cycle application with our construction permit followed by our operating license application. And highlighted on the slide here, in terms of the approvals for the plant, you know, we'll come back before the ACRS again multiple times. What was mentioned earlier, we'll touch on some of the topical reports, so there's other opportunities to come in front of the ACRS, as well.

Part of the reason for the Part 50 process is tied to something that Chris mentioned. You know, we're a part of the ARDP program, so there's an aggressive time frame that we're working under and the two-part Part 50 approach lends itself to starting construction earlier and engaging with the staff earlier to facilitate that time frame.

Currently, if we look at this slide, we are to the left of it, so we're kind of off this slide, you know. So we will get our construction permit application submitted and start this flow path.

Let's see. Topical reports we'll touch on in a minute, and I'll speak to those time lines specifically.

MEMBER MARCH-LEUBA: Just a question. On the construction permit, your intention is to submit it before you really have a finalized design?

MR. SPRENGEL: Yes, yes, yes.

MEMBER MARCH-LEUBA: It would be kind of what I call aspirational, the statements.

MR. SPRENGEL: I don't know if I would consider aspirational, but we will be following on 10

CFR 50.34, speaking to a preliminary state of the design. As was noted, though, our project --

MEMBER MARCH-LEUBA: It will mostly set requirements that you have to meet on the license.

MR. SPRENGEL: In some areas, yes. In some areas, there will be things that will be planned out. Some where testing, certainly, will be ongoing. The structures, after we have the CP submitted and reviewed and issued as a construction permit, you know, our project is set up to start construction.

MEMBER MARCH-LEUBA: From the typical point of view, when you review a CP, you're very disappointed. There is no information there. So I'll focus myself to adjusting that way.

MR. SPRENGEL: Okay.

VICE CHAIR KIRCHNER: I think what Jose is referring to is could you give us just maybe some feeling for what the maturity of the design is when you go into the TP stage? Are you ready, at that point, to write procurement specs or you're still, is the design evolving subsequently?

MR. WILLIAMS: I'd say it's definitely taking the design down beyond plant functional requirements to system requirements. And then the safety analysis is mature enough to ensure that, you know, the plant has enough safety margin in it to start construction, so you'll see a lot of the design basis accidents and even beyond design basis accidents, the PRA. Because we're using licensing modernization program, which Ryan is going to talk about here, PRA is a very, you know, factors very prevalently in the design from the beginning, so there will be a lot of that information available. So it doesn't mean we're done with design, but we have to have established the safety margins of the design.

MEMBER MARCH-LEUBA: So it won't be a complete PRA, but at the CP stage you expect a decent PRA?

MR. WILLIAMS: I do, yes.

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

CHAIR REMPE: You know, I was puzzled with your comment, too, and I guess, you know, we've seen a range of maturities in what we're seeing, and some are very good and, yes, they identify what else is needed, the staff does, and we review it. And then they make changes, and that's wonderful. But it helps, I think, to focus the questions on what you might want to consider that needs to be changed. That's where he's coming from, not that, you know, the way you responded back was a little puzzling to me that you were going to change your focus. But, anyway, I think that will be good to see more detail is the bottom line.

MR. LEVESQUE: And if I could add, this is Chris Levesque, TerraPower CEO. Just a reminder about how well resourced this project is, I mean, thanks to the Department of Energy and the ARDP grant and thanks to our shareholders. I mean, we have 800 engineers working on the project today, getting into considerable design detail, some of which you'll see in the closed session. So there's a lot of rigor and a lot of detail being established today that I think you'll see in the construction permit application.

MEMBER MARCH-LEUBA: I'm glad to hear that.

MR. SPRENGEL: To date, we have had several engagements with the NRC staff to talk about our PRA. It is integral to our use of LMP. And one of the benefits, we talked about PRISM before and how PRISM has served as, you know, part of the foundation of the Natrium design. PRISM had a very well-developed PRA, and we have used that as a really advanced starting point. So we're not starting from nowhere on our PRA, and we've gone further since that time. I do think our PRA is quite advanced at this point, and, of course, we'll go forward with additional steps of peer reviews, and that's kind of going to be going on in parallel to our CPA. But all that to be said that we are moving forward and we are not trying to just submit the very minimum because, at the end of this, we are planning to construct. We are moving right into activities, and so I think our level of detail should be, hopefully, more advanced to meet the needs.

Okay. So licensing modernization project, we are using it. Of course, some of the aspects of it, LB identification, SSC classification, and defense-in-depth, that's just part of the fundamentals of what the LMP process outlines. NEI-18-04, serving the basis of that and the staff's endorsement of it. We're also using Reg Guide 1.232, so recently we've submitted our principal design criteria into the staff January of this year. So that is undergoing staff review at this time.

And then at the bottom there, building off of what was mentioned earlier, we're following the ARCAP and TCAP activities for the content of our application. And so NEI-21-07 is one of the kind of foundational documents amongst the many, many draft guidance documents that are out there. We are continuing to follow those and we'll submit under the ARCAP-TCAP kind of structure of the CPA.

MEMBER PETTI: So I don't want to put words in your mouth, but it sounds like if Part 53 were here today, you've got many of the core elements of Part 53.

MR. SPRENGEL: Sure, yes. I think, yes, Part 53 not being here, I guess we haven't gone that far, but yes.

MEMBER PETTI: But you've certainly got, I'm going to call it the heart of it.

MEMBER MARCH-LEUBA: At ACRS, we fly at a 40,000-foot level and trying to see everything. One thing we emphasize a lot is the LB, the licensing basis event selection. It's human nature to start with light water reactor events and scratch out the ones that aren't going to apply to me. What we emphasize a lot is you should not do that. Start with a wide sheet of paper, we hear white sheet of paper a hundred of times while we're here, and think of what could possibly go wrong in your plan that doesn't apply to light waters because it is very human nature to scratch and don't add. So we've been looking into that.

MR. SPRENGEL: Thank you for that comment.

We haven't started with light water reactor defense, so it is challenging to, you know, to navigate LMP and understand exactly what's a good LB list. But there's still challenges in following the process because it's brand new, but we're using it.

MEMBER MARCH-LEUBA: LMP is great. If you forget an event, it could give you the wrong answer. Bad input, bad answer.

MEMBER PETTI: The other thing, to piggyback on that, is you've mentioned safety margin because this is sort of first of a kind. You've built out the sodium system, but there's a lot of newness here. And for us, being at 40,000 feet, you know, to have the safety margin buried in some appendix or some technical document supporting the PSAR doesn't do us, you know, as trying to get the public confidence up, where you guys can come up and say a margin here, this is the margin there, you know. That is a really easy. You'll start to see how our letters on advanced reactors look. They look very different than our letters on water reactors purposely because we recognize the first-of-a-kind nature, and so we're looking at things in an LMP Part 53 sort of framework, you know, top-down because these are so new, you know.

So I encourage you to make sure that your safety margins are clearly articulated when you come to us. It makes our job easier to represent you guys properly.

MR. SPRENGEL: Thank you. Looking at our topical reports that we submitted, so there's five listed here. Our quality assurance program description has already been approved. We have a revision that's going through staff review currently in the final stages. I guess, actually, we have the draft SEN in hand right now.

George mentioned earlier the energy island interface. That's one that has gone through a staff audit and completed that audit, and so, you know, likely, that would be the next one that would be up for scheduling effort to get in front of the ACRS subcommittee for the Natrium project. I mentioned the principal design criteria here, that was submitted in January of this year. And then the other two are more recently submitted in the last few months, our fuel and control assembly qualification. That's building off of some additional work that we did with the staff and with the Department of Energy on generic advanced fuel qualification, so we used that --

MEMBER PETTI: The NUREG that the staff produced, advanced fuel qualification?

MR. SPRENGEL: No. In this case, the delivery was actually to the Department of Energy.

MEMBER PETTI: Oh, okay. Because you know there's a NUREG-22, I can't remember the number.

MR. SPRENGEL: Yes, yes. Those are, yes, they're interlinked in terms of how they were developed absolutely. The topical report we submitted was specific for TerraPower's fuel and control assembly qualification and also touches on things that Eric mentioned that are kind of next generation of fuel that we have on our radar and is important to our overall project.

And then our emergency planning zone methodology, which is -- one part we'll see on the upcoming reports, as well. There's other pieces that really fit into that EPZ methodology.

MR. BLEY: Before you leave this, Dennis Bley again. I assume you've done something like a PIRT and identified knowledge gaps and things you're working on. If you can say something about that, I'd appreciate it.

Also, it appears you've really looked into the salt usage by the solar plants, and you ought to have pretty good reliability data and the like. I don't see any reason why your tanks and piping systems would be a whole lot different from theirs, except for maybe temperature. Can you talk about that, too.

MR. SPRENGEL: So the first question on PIRTs, we've got, I guess I don't know, I don't want to sound too slang usage here, but we do have lots of PIRTs and in several different topic areas. Recently, I'm trying to think, we met with the staff and presented on our use of PIRTs for our core blockage methodology. We've got some other ones in thermal hydraulic uses. So we've got, you know, PIRTs have been fundamental to how we've looked at what, how we model things and then any gaps that we needed additional testing to support.

MEMBER MARCH-LEUBA: Are those likely to be part of the docket, or will it be only internal documents that the staff would have to order you see them? Can I see them eventually?

MR. SPRENGEL: I don't want to speak for the staff on that. I'm not sure if PIRTs are part of the basics of what we would submit, so certainly they would be available.

MEMBER MARCH-LEUBA: See, the fact is the staff, when they need to see one of your internal documents, they just see it. We at ACRS don't have that. Number one, we compress our review dramatically. And number two, we don't have access to those things. So think about the fact of that. A reading room, one of those famous reading rooms, would help our review.

MEMBER PETTI: Other applicants have done that. It's very effective.

MEMBER BALLINGER: You don't need a PIRT for a construction permit for the salt side, right? I mean, at what point is the boundary, good old fashioned industrial plant where you can build at risk and you don't need a construction permit for that.

MR. SPRENGEL: Yes, I don't think we're doing a PIRT for the energy island. It's just the review of operating experience to understand, you know, taking the technology from a concentrated solar plant that had different requirements on reliability to a nuclear plant that wants a high-capacity factor, is there anything that we need to do or is it good enough? That's the investigation, not really a PIRT.

CHAIR REMPE: But we do have to do an environmental assessment for the energy island.

MEMBER BALLINGER: For anything. What I mean is you don't need a construction permit from the agency to build the salt side, only to a certain point, right?

MR. WILSON: That is correct. This is George Wilson. That is what's in the topical report. It talks about what regulations are applicable or not, so there is a portion of the energy island that would not meet the definition of construction under 50.10, and we would be able to construct it without a limited work authorization.

The salt system we're still having the staff look at specifically for portions of it.

MR. SPRENGEL: I want to check in. Dennis, I think there was a second part to your question. Did we cover it?

MR. BLEY: I think you got it. Oh, yes, I asked have you been able to get pretty good information from the solar folks who use the salt systems, and is there anything really different between your tanks and the things they use?

MR. WERNER: Yes, this I Mark Werner. We've been canvassing the concentrated salt power industry quite a bit, and we've got a couple of SNEs on staff that have direct hands-on experience with the systems. So we've been pulling as much knowledge as we can. Being partnered with the DOE through the ARG project has also been helpful because they've got a number efforts through, I think, NREL that we've been able to kind of attach ourselves to.

And so we are kind of following some of the tank, I'll call them issues that have been cropping up at different plants and feel like we've got a good plan to address the issues that they're currently seeing. I think one thing that we will bring to the table is, you know, we will probably bring a heightened level of quality to the overall project because it is attached to a nuclear plant. I think that will help out for sure.

MR. BLEY: I don't have a clue what temperatures they operate at. Are you a whole lot higher than them? What I'm thinking is is information on system performance and reliability going to be a transferrable from the solar plants over to what you're building?

MR. WERNER: Yes, it will be directly applicable. Our temperatures are very close to how they operate. I think they might have a slightly higher hot, and we're kind of down at a slightly colder cool, but we overlap very well.

MR. BLEY: That's pretty encouraging. So there's nothing really new here.

MR. WILLIAMS: Yes. And the salt that we've chosen for this system is this commercial solar salt, so when you hear about solar salt it's a fairly commercial product, and that's what we're using. You know, a lot of the molten salt reactors use a very advanced salt and highly corrosive and whatnot. But this is a well-known salt, and there's lots of it made. It's a commercial product.

MR. BLEY: I had a question about the physical nature of the salt. You have really hot liquid salt moving through these pipes. If you get some kind of a break or a leak in a pipe, what's the characteristics of that salt as it comes out? I don't know if salt, if it can flash or what happens to it out there. What's the hazards the operators are going to have to know about and worry about?

MR. WILLIAMS: No flashing. It's at atmospheric pressure, so it comes out as a hot liquid and turns into a solid, like that white cake-like substance.

MR. BLEY: So pretty benign as it comes out.

MR. WILLIAMS: Yes, it is.

MR. BLEY: Except if you're touching it.

MR. WILLIAMS: Yes, except it's hot. So I should mention, though, our pipes are atmospheric pressure, but they are high temperature. So, you know, that definitely involves some design consideration because of that.

MEMBER BALLINGER: At 600 C, you're out of the normal stainless steel range. So have you thought about -- the salt people don't use 316 stainless steel. They use other stainless steels.

MR. WILLIAMS: Yes, we won't get up that hot.

MEMBER BALLINGER: Well, it said 600 C.

MR. WILLIAMS: Oh, 600 C, I said stainless steel shows favorable performance with the solar salt up to 600 degrees C. But our core outlet temperature is 510, so nothing is going to be up that hot.

MEMBER BALLINGER: I've got 475 stuck in my mind. Certainly, for Section III.

MR. WILLIAMS: Yes.

MEMBER BALLINGER: Okay, all right. I've got to go back and look and see.

MEMBER PETTI: So can you do it in the traditional nuclear part of the code, or do you got to go div 5?

MR. WILLIAMS: We go to div 5.

MEMBER PETTI: You do go to --

MR. WILLIAMS: Yes, we do.

MR. SPRENGEL: Looking ahead at our

planned topical reports, many of them listed here. We continue to work with the staff on these submittals and getting them to review. This month, we're targeting the HFE program plan document, as well as volcanic hazard assessment. And looking ahead into summer, we've got source term, we talked about relation to our EPZ methodology.

Later in the fall, our DBA transient methodology, partial flow blockage, and our code usage. And then in the winter, digital I&C and fuel handling I&C. So lots of activity in terms of getting these into the NRC staff's hands for review this year.

MEMBER MARCH-LEUBA: I see you have a topical on stability. Are you worried about oscillations in the core, or is it thermal hydraulic -- what in stability are you worried about?

MR. WILLIAMS: It's not VNOM transients; I know that. But probably need to call back to the home office.

MR. SPRENGEL: So the stability methodology is, generally, it's describing how the core functions in steady state, so it's kind of a stable reactor.

MEMBER MARCH-LEUBA: Not that it's unstable.

MR. WILLIAMS: Right.

MR. SPRENGEL: Okay. Looking at, we did submit our Natrium engagement plan back in June of 2021, coming up on two years of pre-application engagement with the staff, 39 meetings to date. We have seen benefits and have a good working relationship with the NRC staff. And reiterating here our PSAR content is being developed consistent with the ARCAP guidance, so the structure will be different under the ARCAP guidance from NUREG-0800.

VICE CHAIR KIRCHNER: How do you view that, Ryan? Is that just a mapping exercise, or are you the first one going through the wicket, or both?

MR. SPRENGEL: Yes, both is probably a better way. The guidance in the form of the groups who have done all the guidance between ARCAP and TCAP and it's been spread out into, you know, who kind of has primary, who has the primary lead on developing that guidance, so it's spread out in many documents, and I think all of it is still draft at this point. So it does lay out the structure of what goes where.

I guess I'll give an example of where the first time, you know, evolution has come into play. Some of the structure we've modified and we've worked with the staff and, you know, industry stakeholders in terms of some modification of where things might fall just to make sense, primarily from a reviewer and packaging standpoint, and have proposed that that's our path forward, and I think some of that is being incorporated, you know, as the draft finalization.

We are also going back and we have reviewed regulations and NUREG-0800, the SRP, for any applicability and any kind of gaps that might be there, and we continue to engage with the staff. You know, when we identify something that we see in expectation from NUREG-0800 but we don't see described kind of one way or the other or maybe lightly in the ARCAP guidance, we reach back out and are looking for clarification on the staff of is that intentional that it was actually removed. Most often, that's not the case. And so then we take that on as an additive of --

VICE CHAIR KIRCHNER: That would be my concern because you're doing it as a construction permit, and that's guided by 50.30 and its requirements. ARCAP is coming along later. That's why I said the first order, the mapping exercise. But then whether the staff -- it's something we'll ask the staff, not, you know, are they ready to map NUREG-0800 or some version of NUREG-0800 to the ARCAP guidance, so that there's not a lot of -- what I would worry about is not so much that you don't have all the content there, but there's this, like I said, a mapping exercise, and so that the review is efficient for both parties.

MR. SPRENGEL: Absolutely. Yes, we are not, we're following the guidance, but we're also doing our due diligence to review the regulations and other sources of kind of, you know, NUREG-0800 is just had more time to develop and be complete and comprehensive, and so we're using the regulation, as well as guidance, to inform and kind of cross-check all those things on our side. I agree it would probably be a fair question for the staff of how they're viewing that, as well.

MEMBER BALLINGER: I haven't lost my mind. I'm looking at Table HAA-1130-1 in Division 5, and the upper temperature limit for stainless steel is 425. So I don't know what section we're working to, but Section VIII allows higher temperatures and allows higher allowables. But Division 5, Section III, Division 5, there's a limit there. I saw 600 C on one of your slides, and so maybe I ought to check this.

MR. SPRENGEL: Yes, I'll have to check back on the code applicability. I just meant we don't have anything up at that 600 degree C temperature because the core outlet is at 510. We do have it above 425.

MEMBER BALLINGER: I mean, Division 5 is 640 pages long, and I'm sure you probably missed something.

MS. NEIDER: We'll bring that back to our team and evaluate it.

VICE CHAIR KIRCHNER: Members, any questions before we take comments from the public? Okay.

MEMBER SUNSERI: Just a question, I guess. I heard a lot of the members talk about Part 53, ARCAP, these kind of things. None of that is approved for us, and they won't be licensing to that, so I don't want to send you all a message that we're holding you accountable to Part 53. Are we? We're not, right? But all we've talked about, we've talked several times about that, so I just want to be clear that's not the standard.

VICE CHAIR KIRCHNER: That's where I was going, Matt, that, obviously, they have to use Part 50. That's what they're applying for, construction permit under 50.

MEMBER BROWN: The point is don't ask

questions on the other ones.

MEMBER SUNSERI: I mean, it was just sounding to me like it could get confusing to the applicant is all.

MEMBER MARCH-LEUBA: I'm going to make my typical talk at the end of this presentation based on this comment that we're asking a lot of questions. We don't have any problem with your design. We like your design, and you're doing a great job, and we like this stage of the project. When we want to start scratching on the surface, then we ask questions that you won't like. But right now, thank you for this. Don't take anything we said as meaning anything detrimental.

VICE CHAIR KIRCHNER: And these are comments by individual members at this juncture. This is an informational briefing for us, not a critique of the design.

Okay. I'd like to open the floor to any comments from the public. If you're participating online, please mute your microphone, state your name, identify your affiliation if relevant, and make your comment, please.

MR. MOORE: This is Scott Moore. You may need to press *6.

VICE CHAIR KIRCHNER: Yes, let me see here. We see a hand is up out there. Oh, Ed, yes, go ahead, Ed. Unmute your mike, please, and make your comment.

MR. LYMAN: Yes. I thank you. This is Ed Lyman from the Union of Concerned Scientists. I'd just like to point out that we have some serious safety concerns with fast reactors in general and this design in particular. And in the discussion so far in the open session, you only touched on some of the relevant aspects. And what's frustrating is Natrium does not seem to be transparent. When they talk about the difference between sodium-cooled fast reactors and light water reactors, they rarely mention that the time scale for transients is so much shorter in fast reactors that that is a significant safety flaw. And they emphasize the temperature difference between the operating temperature and the boiling point of sodium, but they don't talk about those transients that could lead to a very rapid increase in coolant temperature. And, yes, this reactor does have a positive void coefficient, and, if it does get to a boiling crisis, you could have a very severe reactivity transient.

And the absence of a physical containment is a potential significant safety vulnerability they also don't talk about. I know the core assembly events, which I'm sure everyone knows, was a big factor in the Clinch River and FFTF licensing and, as far as I'm concerned, has not been resolved. And so the lack of a strong physical containment is, again, I think a significant open question for this design.

So I urge you to explore these questions both in the closed session and also in future open sessions, and that's my comment. Thank you.

VICE CHAIR KIRCHNER: Thank you, Ed. Any other members of the public wish to make a comment? Hearing none, then what we will do now is close this Teams' link and go into closed session. Do we need to give any other formalities?

CHAIR REMPE: I'd like to suggest that we break until 3:15 to allow the transition to the closed session occur. Also, this is the last that we'll have the open session open today, and so, for those members of the public who want to tune into tomorrow, we'll resume at this link tomorrow at 8:30. Okay.

Thank you. And we are recessed.

(Whereupon, the above-entitled matter went off the record at 2:54 p.m.)

TerraPower

NATRIUM

Plant and Licensing Strategy Overview

a TerraPower & GE-Hitachi technology

NAT-3292

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

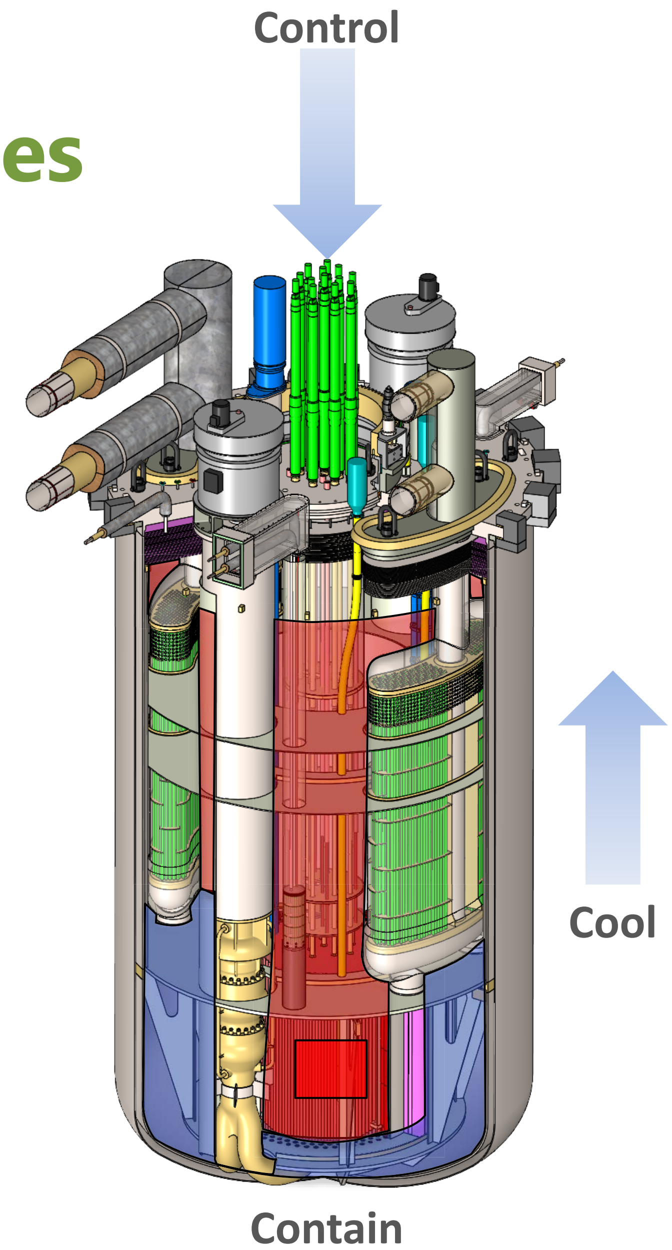
Purpose

- Provide an overview of the Natrium^{TM} plant design and operation, including the Nuclear Island and Energy Island.
- Describe the innovative features and related research and development activities.
- Discuss the licensing strategy for the Natrium advanced reactor.

Please note, the design is not final. There could be changes to systems, components, plant layout, etc. as the design progresses.

Natrium Safety Features

- Pool-type Metal Fuel SFR with Molten Salt Energy Island
  - Metallic fuel and sodium have high compatibility
  - No sodium-water reaction in steam generator
  - Large thermal inertia enables simplified response to abnormal events
- Simplified Response to Abnormal Events
  - Reliable reactor shutdown
  - Transition to coolant natural circulation
  - Indefinite passive emergency decay heat removal
  - Low pressure functional containment
  - No reliance on Energy Island for safety functions
- No Safety-Related Operator Actions or AC power
- Technology Based on U.S. SFR Experience
  - EBR-I, EBR-II, FFTF, TREAT
  - SFR inherent safety characteristics demonstrated through testing in EBR-II and FFTF



Control

- Motor-driven control rod runback and scram follow
- Gravity-driven control rod scram
- Inherently stable with increased power or temperature

Cool

- In-vessel primary sodium heat transport (limited penetrations)
- Intermediate air cooling natural draft flow
- Reactor air cooling natural draft flow – always on

Contain

- Low primary and secondary pressure
- Sodium affinity for radionuclides
- Multiple radionuclides retention boundaries

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Key Differences from Light-Water Reactors

Leverage inherent features:

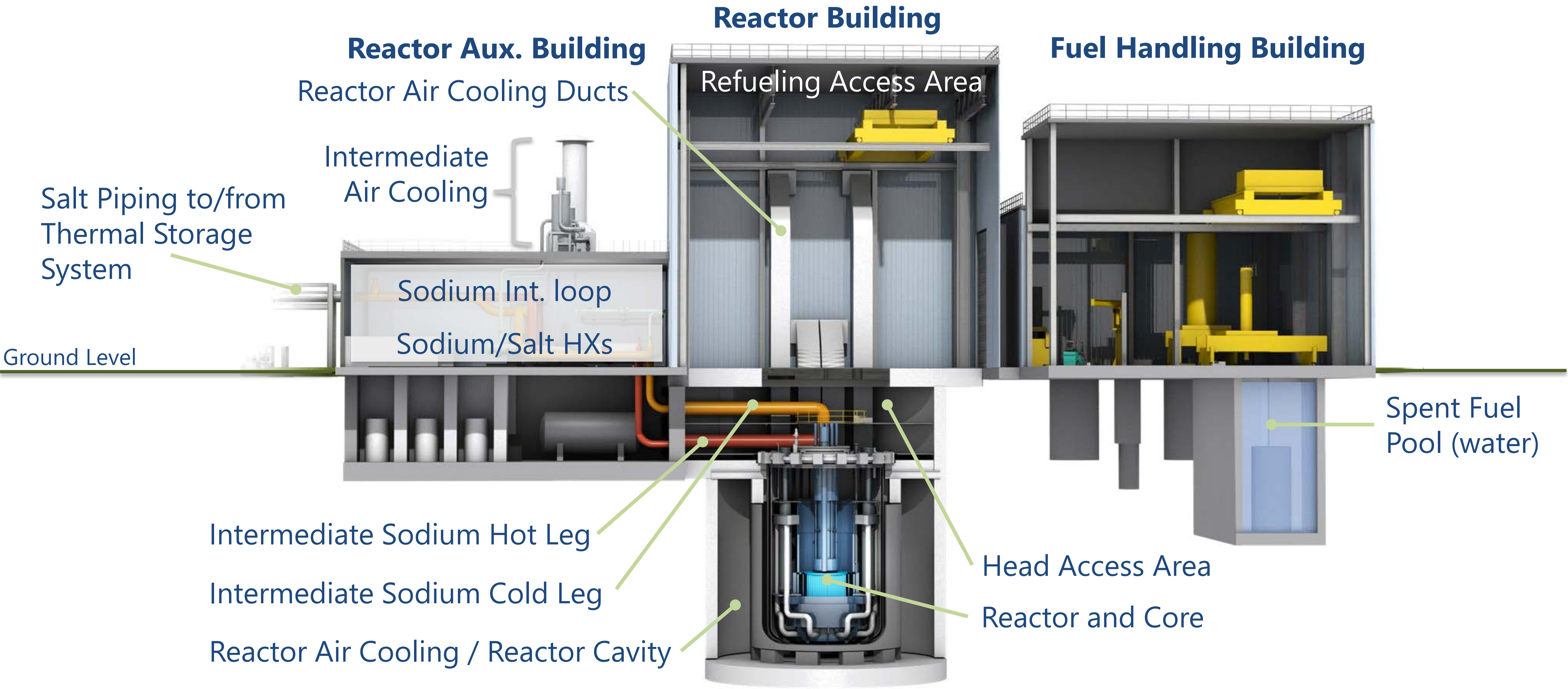
- Compact systems, less “nuclear sprawl”
- Low pressure
- Efficient heat transfer
- Pool design with large coolant inventory
- Modularity
- Parallel construction
- Emergency Planning Zone reduced

NATRIUM

Single Unit Site

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

TerraPower
A Nuclear Innovation Company



Reactor Aux. Building

Reactor Building

Fuel Handling Building

Refueling Access Area

Reactor Air Cooling Ducts

Intermediate Air Cooling

Salt Piping to/from Thermal Storage System

Sodium Int. loop

Sodium/Salt HXs

Ground Level

Spent Fuel Pool (water)

Intermediate Sodium Hot Leg

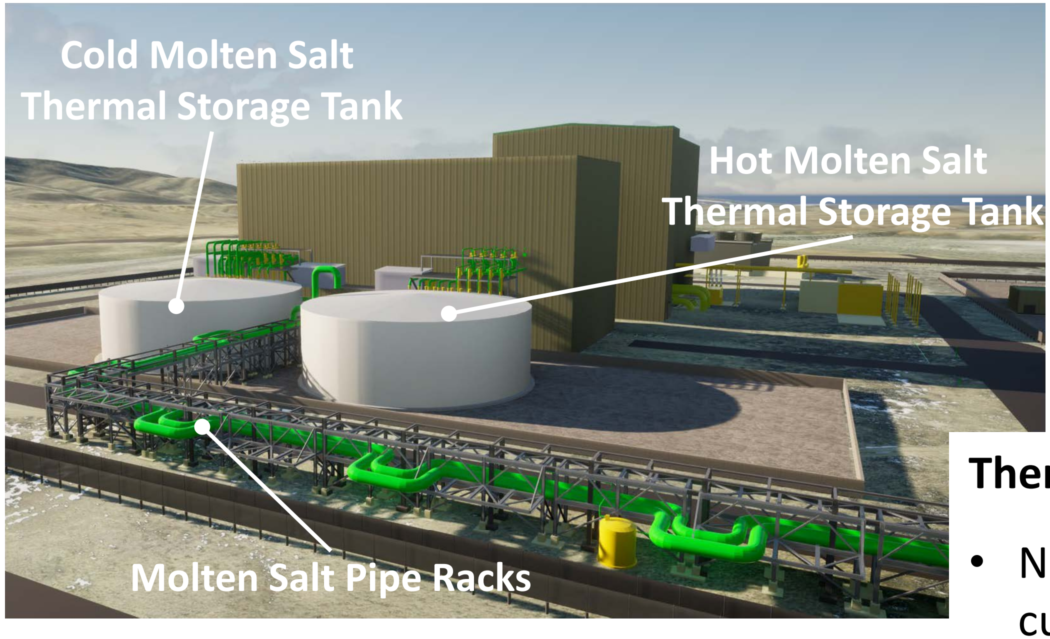
Intermediate Sodium Cold Leg

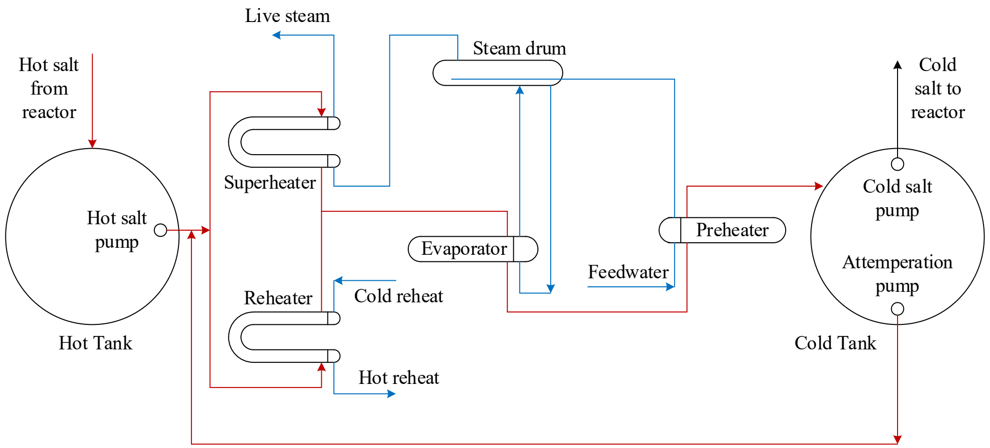
Reactor Air Cooling / Reactor Cavity

Head Access Area

Reactor and Core

Energy Island Thermal Storage





Thermal Storage

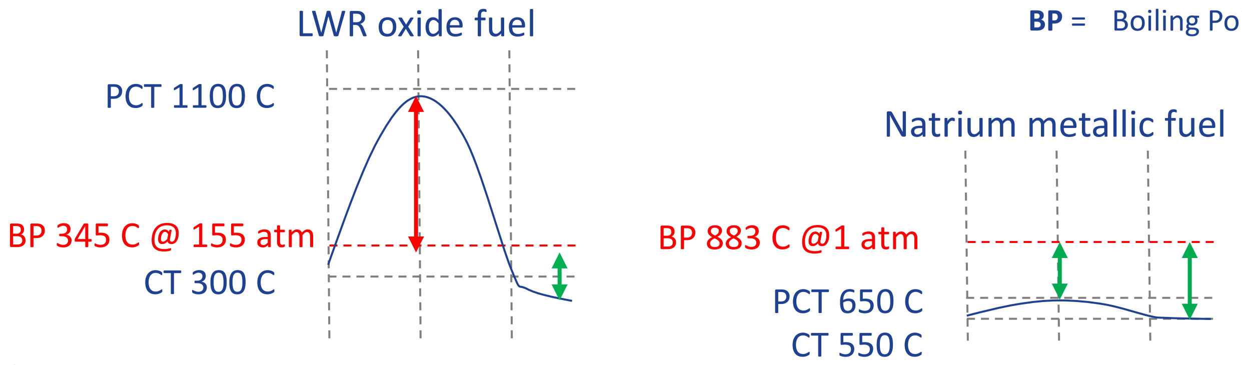
- Number of tanks based on customer’s energy need
- Steam generator trains based on size of turbines
- Turbine size based on customer’s power need

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Benefits of Sodium Coolant

- High heat capacity – moderate flow rate and easy decay heat removal
- High heat transfer – small core and easy decay heat removal
- High boiling point 883°C (1,621°F) – atmospheric pressure
- Low melting point 98°C (208°F) - practical
- Density similar to water
- Lack of corrosion
- Limited auxiliaries
- Sodium inventory - 800 m^3 in reactor

PCT = Peak Central Temperature
CT = Cladding Temperature
BP = Boiling Point *(atmospheric)*



Sodium in Liquid and Solid States

Molten Salt for Energy Storage

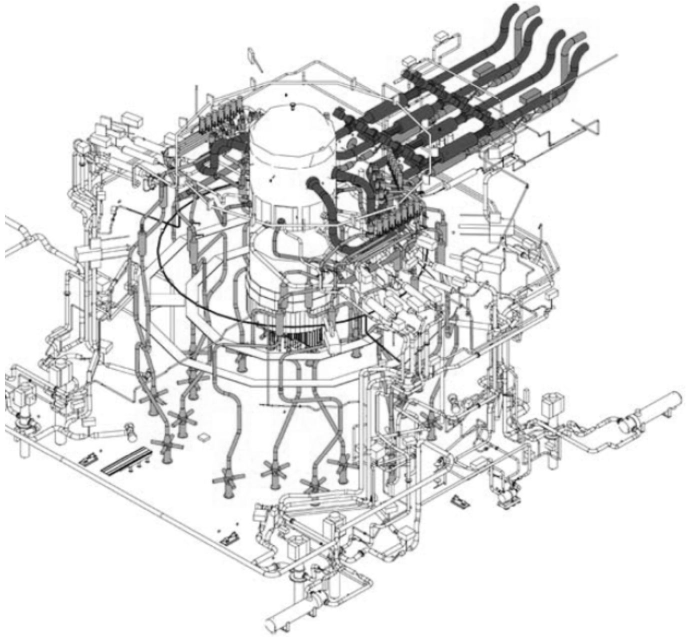
- 60 $NaNO_3$- 40 KNO_3
- Molten salt inventory - 30,000 tons
- Gross tank energy storage capacity with Type 1 fuel – 1,971 MWh

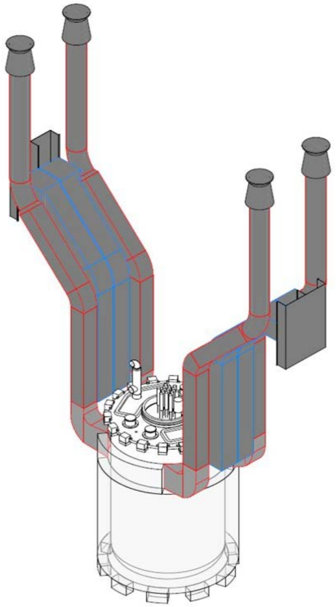
Benefits of using molten salt:

  - Long design life with negligible performance degradation
  - Temperature range 238 – 621°C (460 – 1,149 °F)
  - High thermal energy storage efficiency (~99%)
  - Readily available due to its common use for heat storage and in solar plants
  - Relatively low levelized cost of energy at grid scale compared to battery storage

Salt in Solid and Molten States

Example of Simplified Nuclear Systems





LWR Emergency Core Cooling

- 2600+ ASME Sect. III Pipe Welds
- High Pressure Injection (1000+ PSI)
- Large Water Inventory Requirements
- Active Valve and Pump Operation
- Multiple Trains and Sub-systems

Natrium Reactor Air Cooling

- Zero ASME Sect. III Pipe Welds
- Atmospheric Pressure (<1 PSI)
- Unlimited Air-Cooled Heat Sink Supply
- Fully Passive (Always in Operation)
- Singular Rugged System

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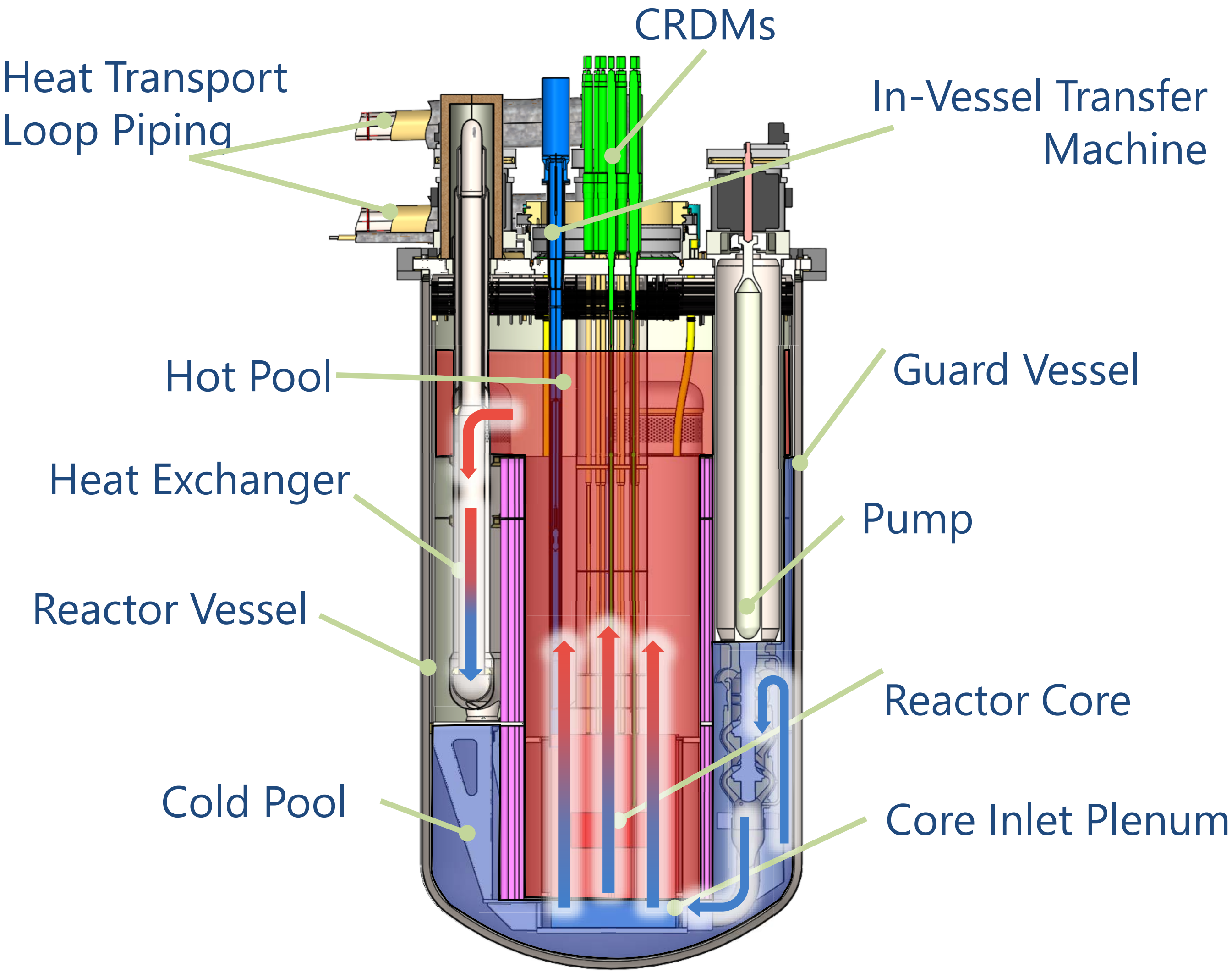
Key Features of Reactor Equipment Design



- Pool Type Integral Reactor
- Large Volume of Sodium Coolant
- Atmospheric Pressure
- Separation of Hot and Cold Pool Regions
- Mechanical Pumps
- In-vessel Refueling

Heat Removal

Normal

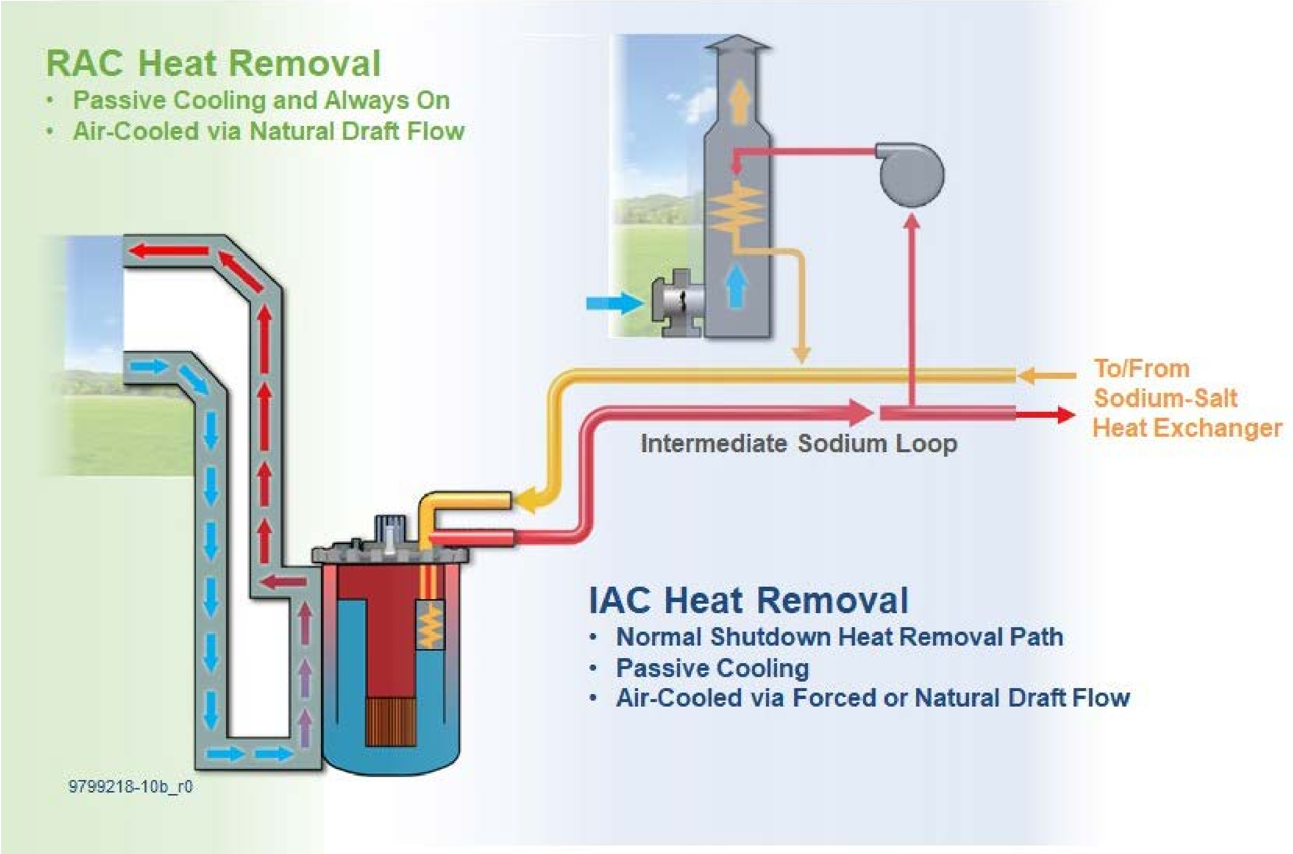
- Intermediate Air Cooling
  - Normal Shutdown Heat Removal
  - Forced Flow

Passive

- Intermediate Air-Cooling
  - Non-Safety-Related Heat Removal
  - Natural Draft Flow

Inherent

- Reactor Air Cooling
  - Safety-Related Heat Removal
  - Natural Draft Flow – Always On



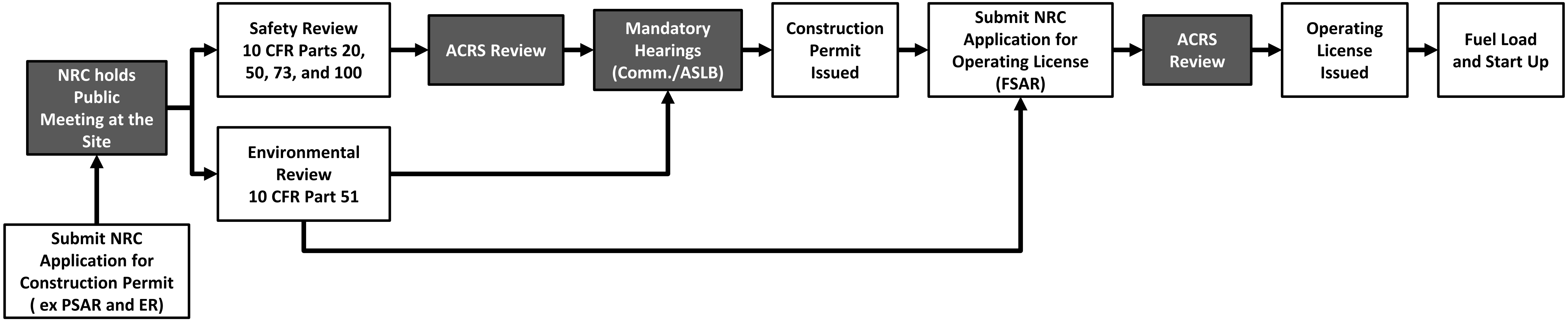
RAC Heat Removal

- Passive Cooling and Always On
- Air-Cooled via Natural Draft Flow

IAC Heat Removal

- Normal Shutdown Heat Removal Path
- Passive Cooling
- Air-Cooled via Forced or Natural Draft Flow

Overview of the 10 CFR Part 50 Licensing Process for the Natrium Advanced Reactor



Submit NRC Application for Construction Permit (ex PSAR and ER)

NRC holds Public Meeting at the Site

Safety Review 10 CFR Parts 20, 50, 73, and 100

Environmental Review 10 CFR Part 51

ACRS Review

Mandatory Hearings (Comm./ASLB)

Construction Permit Issued

Submit NRC Application for Operating License (FSAR)

ACRS Review

Operating License Issued

Fuel Load and Start Up

Key:

■ Opportunity for Public Participation

□ Milestone Activity

Proposed Application of LMP

- Use of LMP for the Natrium design:
  - Regulatory Guide 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors”
  - NEI 18-04, “Risk-Informed Performance-Based Technology Guidance for Non-Light Water Reactors”
  - LMP analysis, including LBE selection, plant-level SSC classification (input to system-level SSC classification) and evaluation of defense-in-depth adequacy.

- NEI 21-07, “Technology Inclusive Guidance for Non-Light Water Reactors” describes SAR content for advanced reactors using NEI 18-04

Topical Reports Submitted

- Quality Assurance Program Description
- Natrium Nuclear Island and Energy Island Interface
- Principal Design Criteria for the Natrium Advanced Reactor
- TerraPower's Fuel and Control Assembly Qualification
- Emergency Planning Zone Methodology

Topical Reports Planned

- HFE Program Plan and Methodologies
- Volcanic Hazards Assessment
- Mechanistic Source Term Methodology
- DBA Transient Methodology (In-Vessel without Release)
- Radiological Release Consequences Methodology
- Partial Flow Blockage Methodology
- Engineering Computer Codes for the Natrium Reactor
- Reactor Stability Methodology
- DBA Transient Methodology (In-Vessel with Release, Ex-Vessel with and without Release)
- Defense-in-Depth and Diversity I&C Strategy
- Digital I&C (Architecture and Design)
- Fuel Handling Instrumentation and Control

Path Forward

- Regulatory Engagement Plan – June 2021
- 39 Pre-application meetings held to date
- PSAR content is being developed consistent with draft Advanced Reactor Content of Application Project (ARCAP) guidance

Questions?

Acronym List

ACRS – Advisory Committee on Reactor Safeguards
ARCAP – Advanced Reactor Content of Application Project
ASLB – Atomic Safety and Licensing Board
ASME – American Society of Mechanical Engineers
BP – boiling point
CFR – *Code of Federal Regulations*
CRDM – control rod drive mechanism
CT – cladding temperature
DBA – design basis accident
EBR – Experimental Breeder Reactor
ER – Environmental Report
FFTF – Fast Flux Test Facility
FSAR – Final Safety Analysis Report
HFE – human factors engineering
HX – heat exchanger
I&C – instrumentation and control
IAC – intermediate air cooling system
LBE – licensing basis event
LMP – Licensing Modernization Project
LWR – light-water reactor
MWh – megawatt-hour
NEI – Nuclear Energy Institute
NRC – U.S. Nuclear Regulatory Commission
PCT – peak cladding temperature
PSAR – Preliminary Safety Analysis Report
PSI – pounds per square inch
RAC – reactor air cooling system
SAR – Safety Analysis Report
SFR – sodium-cooled fast reactor
SSC – structure, system, and component
TREAT – Transient Reactor Test Facility

ENCLOSURE 3

“Plant Overview”
Presentation Material – Closed Meeting

Non-Proprietary (Public)

TerraPower

NATRIUM

Plant Overview

a TerraPower & GE-Hitachi technology

NAT-3293

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

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Portions of this presentation are considered proprietary and TerraPower, LLC requests it be withheld from public disclosure under the provisions of 10 CFR 2.390(a)(4).

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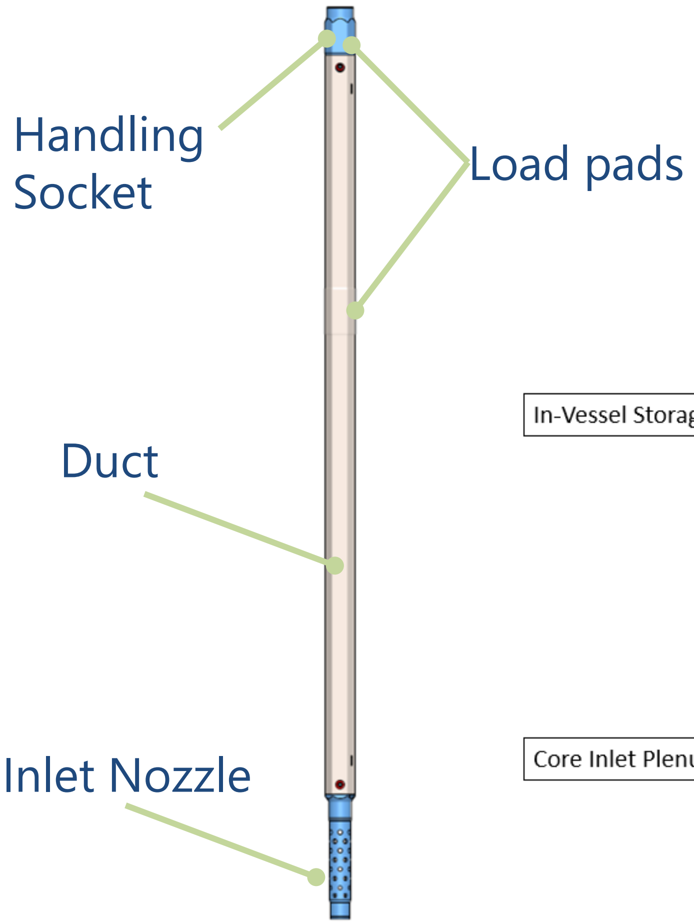
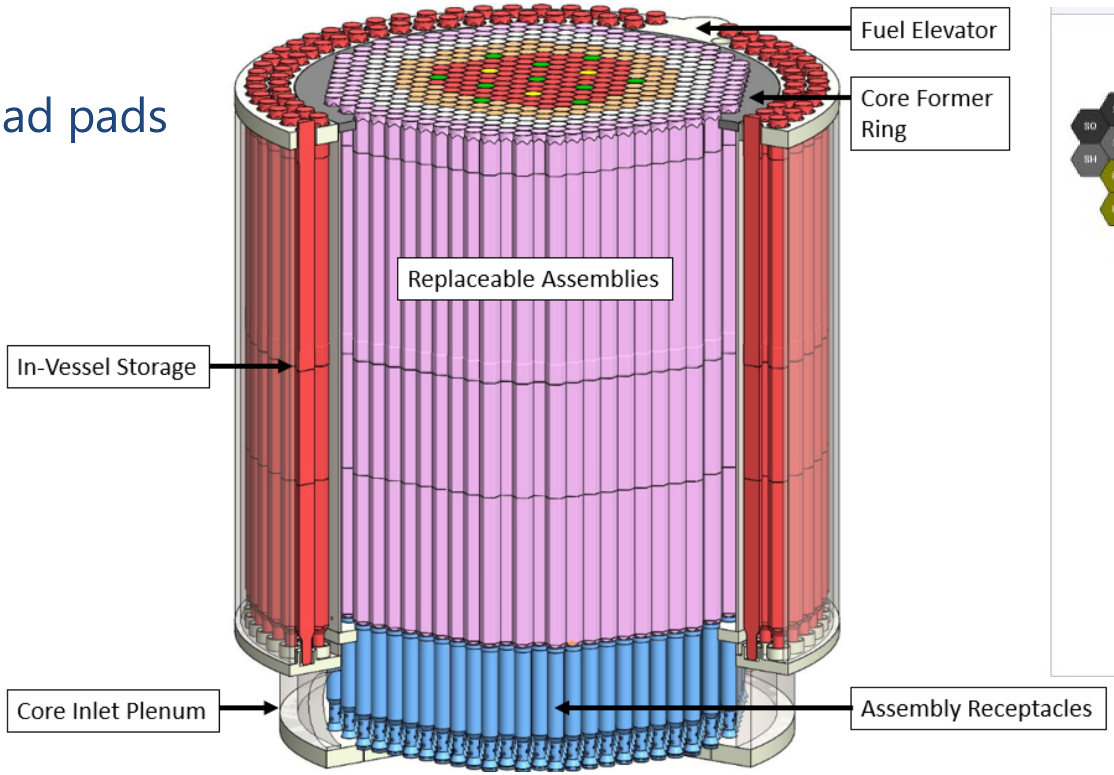
Topics

- Reactor Core Design
- Reactor Equipment
- Refueling Equipment
- Heat Removal Systems
- Sodium Leak Protection/Mitigation
- Functional Containment Strategy
- Source Term and Emergency Planning Zone Strategy
- Energy Island Systems
- Reactor Protection System Parameters
- Representative Transients:
  - Basic SCRAM Design Transient
  - Basic Runback Design Transient
  - Uncontrolled Rod Withdrawal Design Basis Accident (DBA)
  - Loss of Offsite Power DBA

Reactor Core Overview

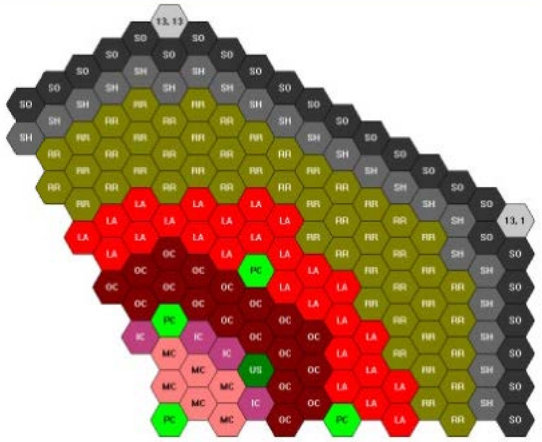
Core Assembly





Reactor Core

Reactor Core Map



PC	Primary Control
US	Standby Control
MC	Middle Core
IC	Inner Core
OC	Outer Core
LA	Feed Fuel
RR	Radial Reflector
SH	Radial Shield
SO	Outer Shield Assembly

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Reactivity Feedback Mechanisms of SFRs

- Doppler feedback: Effect of changes in neutron fission and absorption cross sections due to Doppler broadening
  - – Negative at temperatures above normal
- Core radial expansion: Due to thermal expansion and irradiation-induced swelling
  - – Negative at temperatures above normal due to enhanced leakage and core locked
- Fuel axial expansion: Effect of thermal expansion and transient swelling of especially the metallic fuels (and cladding)
  - – Negative at temperatures above normal due to reduced number density of fissionable isotopes
- Coolant density and void worth: Effect of changes in coolant density at elevated temperatures
  - – Can be positive due to reduced sodium moderation/absorption, or negative due to enhanced neutron leakage
- Control rod driveline expansion: Due to difference in thermal expansion of control rod driveline and reactor vessel
  - – Can be positive or negative depending on expansion relative to reactor vessel expansion

Fuel Types

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Pin Strip Layer Assembly

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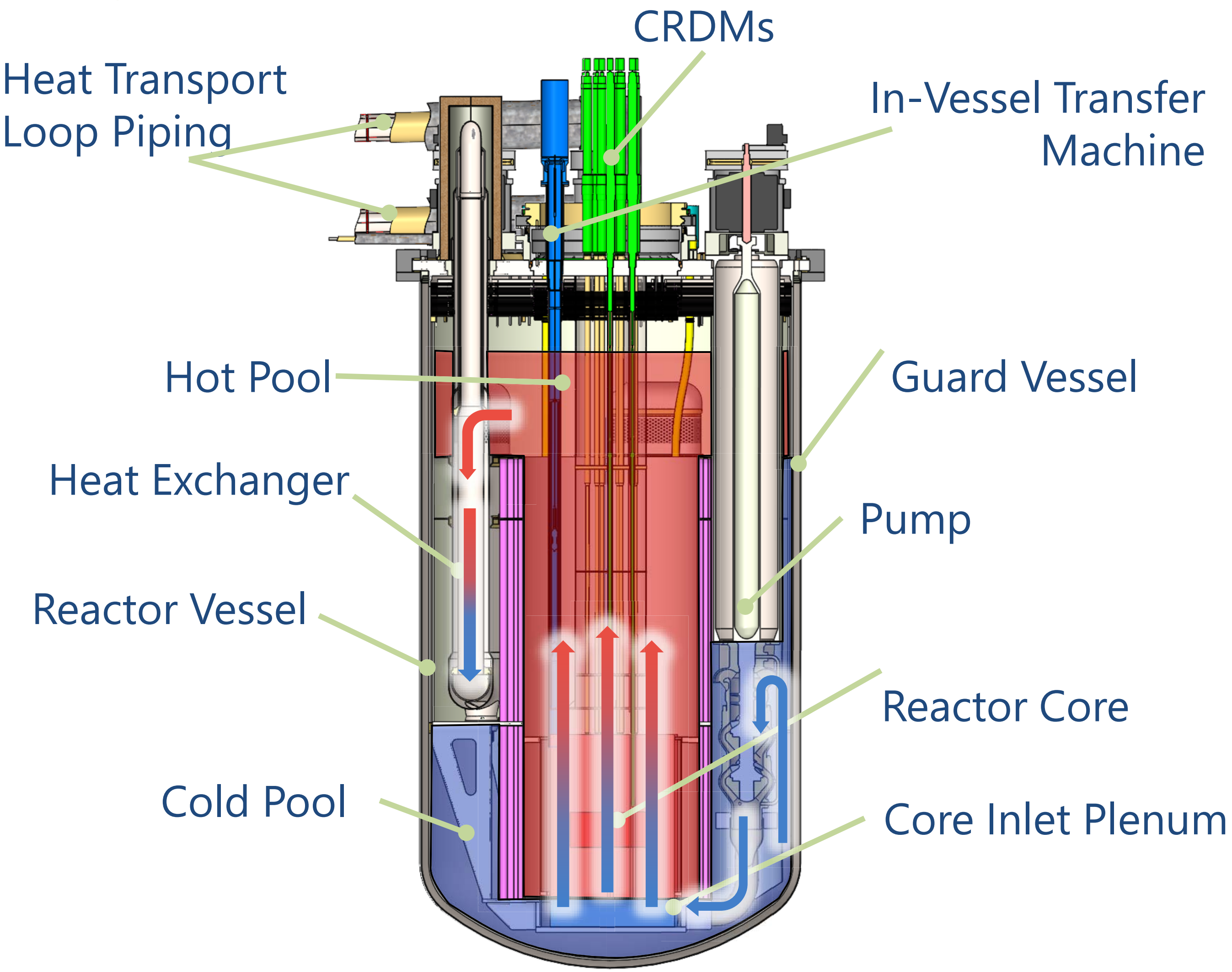
Key Features of Reactor Equipment Design



- Pool Type Integral Reactor
- Large Volume of Sodium Coolant
- Atmospheric Pressure
- Separation of Hot and Cold Pool Regions
- Mechanical Pumps
- In-vessel Refueling

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In-Vessel Refueling

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Refueling Equipment Key Features:

- Remote, in-vessel equipment
- Drives located on head
- Access to all in-vessel locations
- All core components have same interface
- Multiple degrees of freedom
- ***New Fuel***
    - Into reactor vessel (RV) using lift
    - Into core using in-vessel transfer machine
- ***Spent Fuel***
    - From Core to in-vessel storage
    - From storage to lift
    - Leave RV using lift

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In-Vessel Transfer Machine Operations

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Ex-Vessel Fuel Handling Process Overview

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Ex-Vessel Fuel Handling Process Equipment

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Intermediate Heat Transport Loop

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Key Equipment:

- Sodium/Salt Heat Exchanger (HX)
  - Nitrate Salt → Sodium
- Intermediate Pumps
  - pumps similar to primary sodium pump (PSP)
- Intermediate Air Cooling
  - Provide local heat rejection during start up and shutdown

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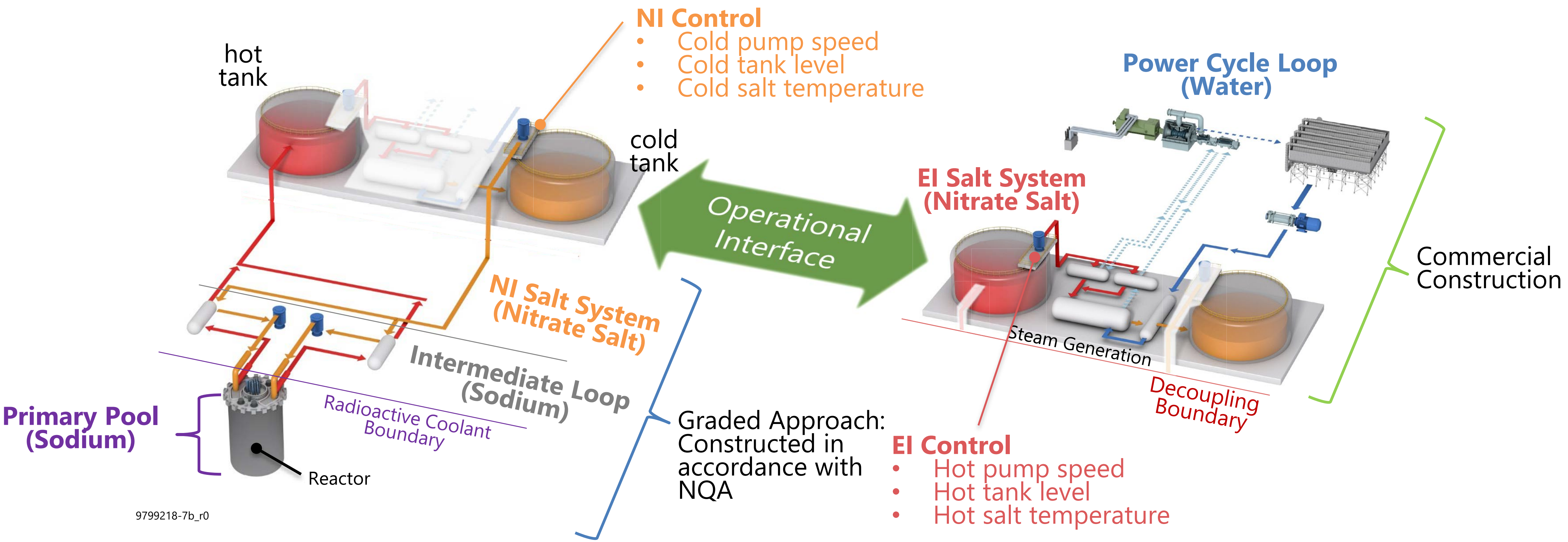
Heat Transport Architecture



NI Control
- Cold pump speed
- Cold tank level
- Cold salt temperature

Graded Approach: Constructed in accordance with NQA

EI Control
- Hot pump speed
- Hot tank level
- Hot salt temperature

Reactor Air Cooling

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Heat Removal

Normal

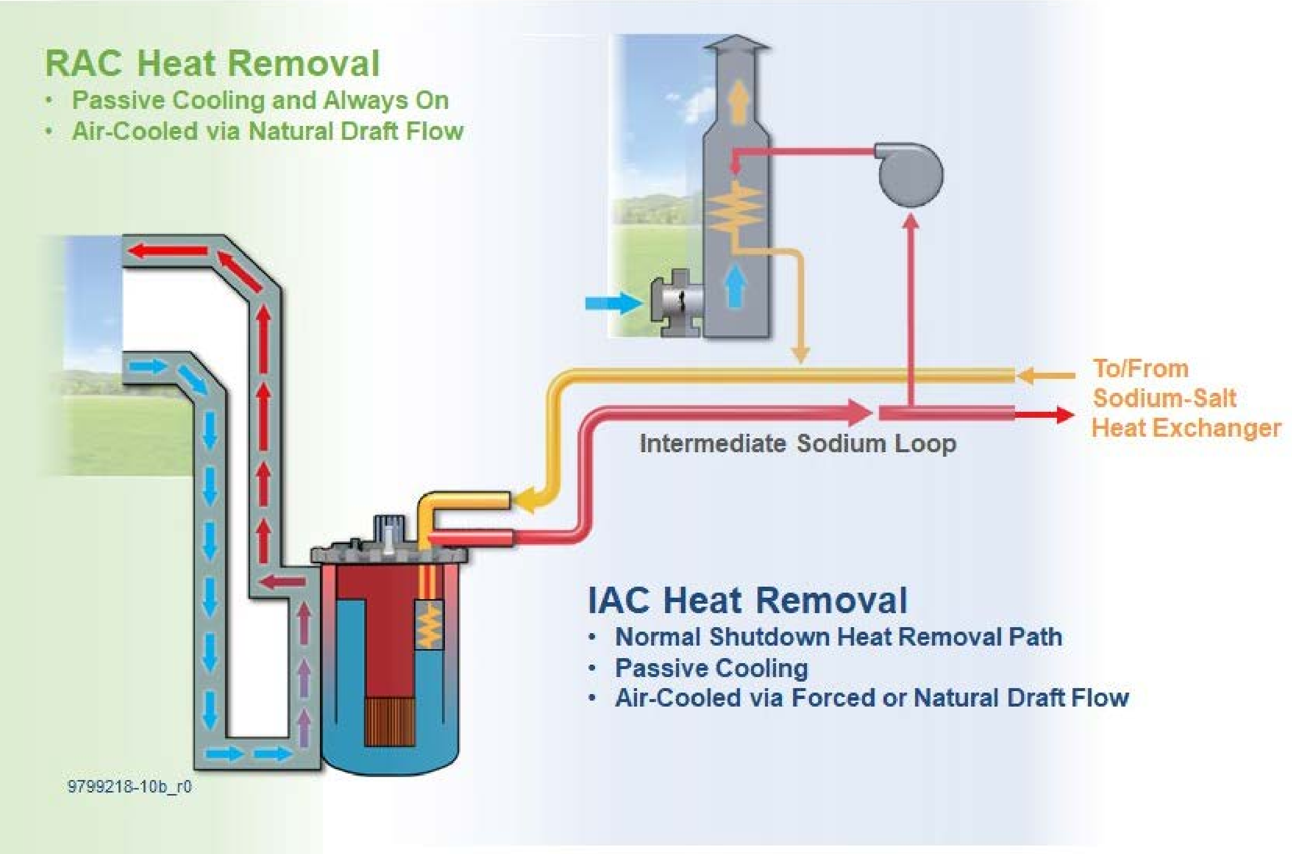
- Intermediate Air Cooling
  - Normal Shutdown Heat Removal
  - Forced Flow

Passive

- Intermediate Air-Cooling
  - Non-Safety-Related Heat Removal
  - Natural Draft Flow

Inherent

- Reactor Air Cooling
  - Safety-Related Heat Removal
  - Natural Draft Flow – Always On



RAC Heat Removal

- Passive Cooling and Always On
- Air-Cooled via Natural Draft Flow

IAC Heat Removal

- Normal Shutdown Heat Removal Path
- Passive Cooling
- Air-Cooled via Forced or Natural Draft Flow

Sodium Leak and Fire Protection

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Prevention & Lessons Learned

- Significantly reduced quantity of sodium piping
- Leak Jacketing / Guard piping
- Remove sodium to steam interface

- **Mitigation features:**
  - Reactor vessel surrounded by guard vessel
    - Inerted
    - Leak and fire detection
  - Reactor Head / Head Access Area
    - Steel lined cells or leak jacketing
    - Inerted
    - Leak and fire detection
    - Leak protection to not compromise SR functions
  - Cells containing Intermediate sodium will have
    - Catch pans with suppression plates at critical areas
    - Leak and fire detection

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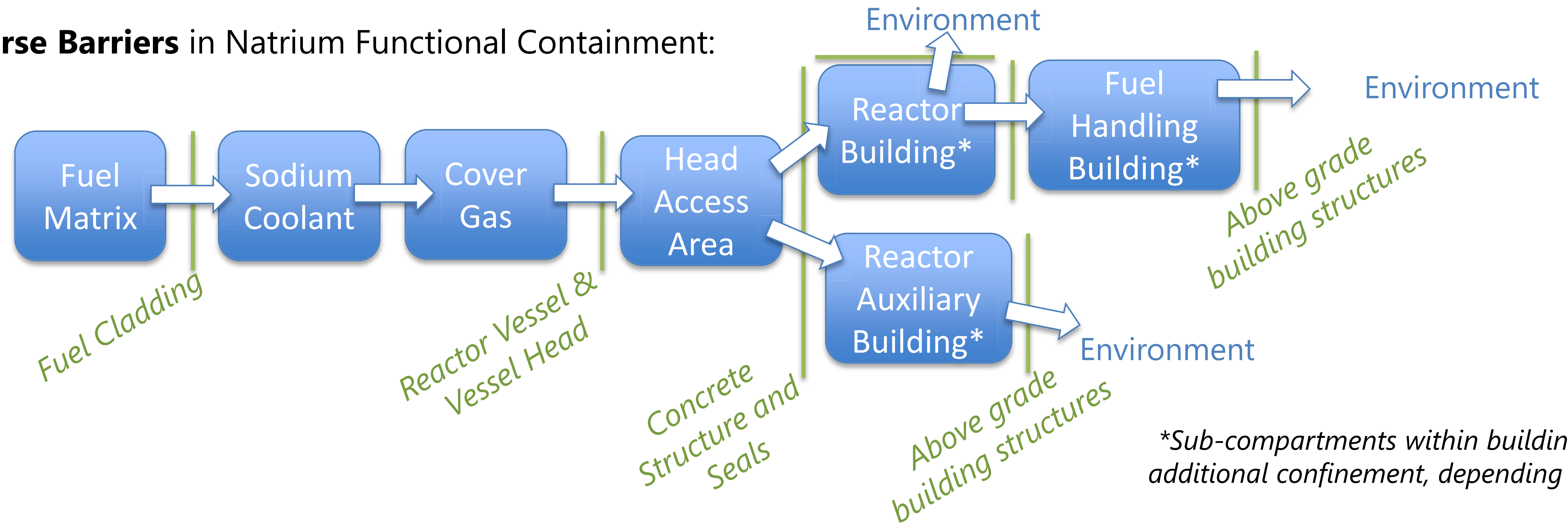
Functional Containment – Diverse Barriers

The Natrium™ design is well suited for Functional Containment:

- **Low** operating pressures & **large margin** to sodium boiling
- **Low** differential pressure between vessel and compartments
- **High** conductivity coolant & **passive** emergency core cooling
- Design precludes any consequential Loss of Coolant Accident (LOCA)

Diverse Barriers in Natrium Functional Containment:



*Sub-compartments within building provide additional confinement, depending on event.

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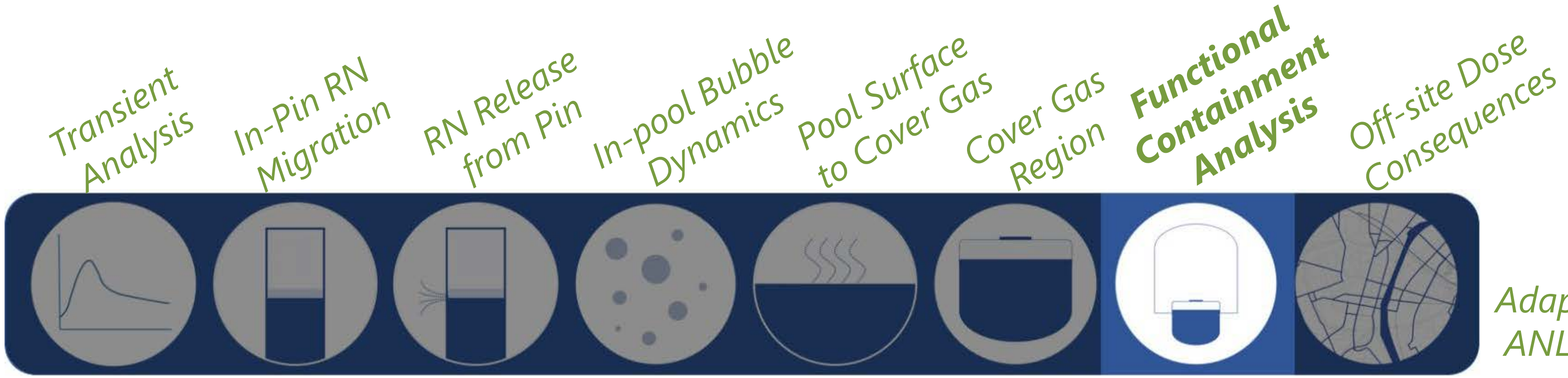
Functional Containment - Analysis and Source Term

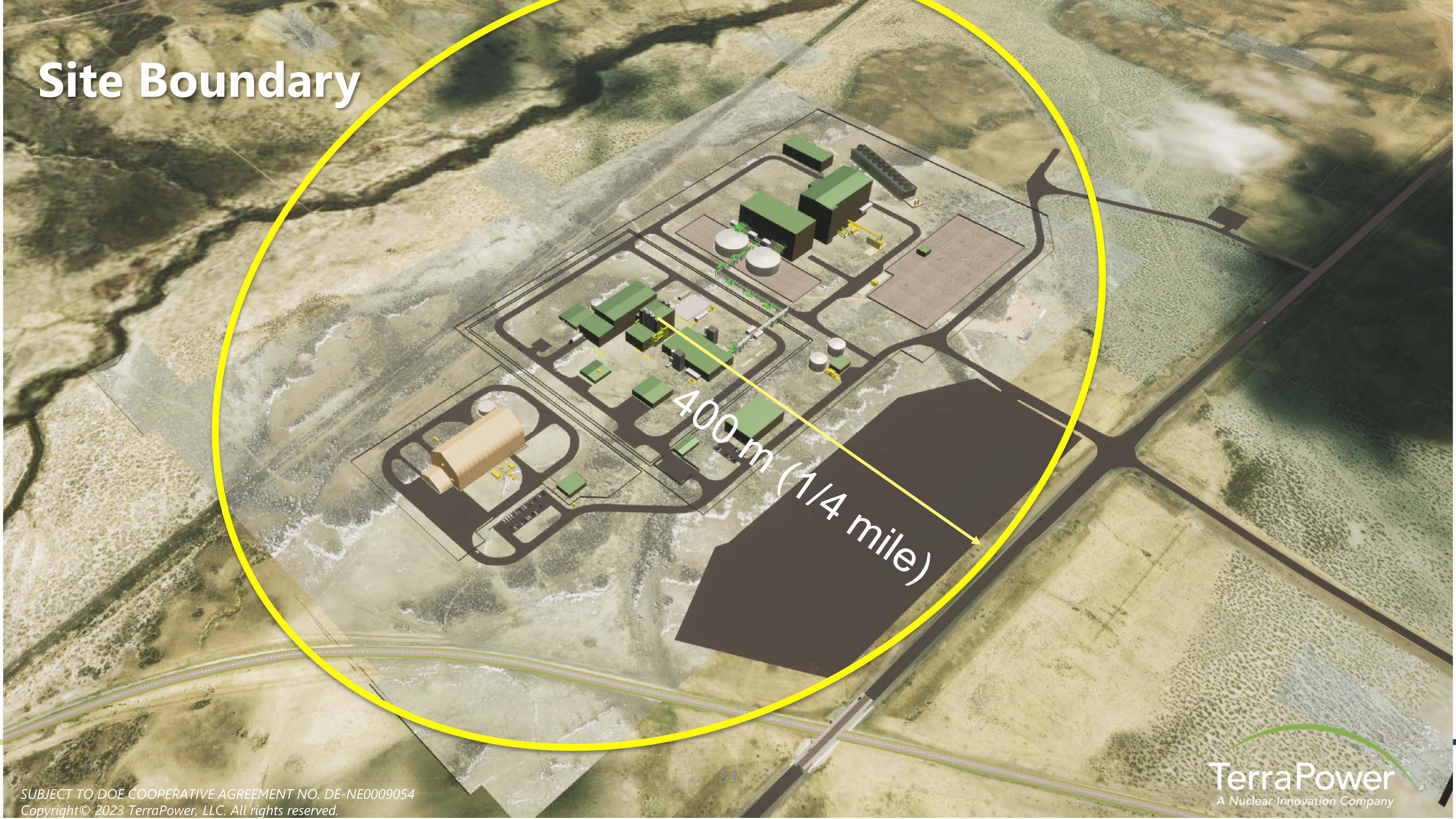
- Quantify compartment to compartment leakage (P/T dependent).
- Assess aerosol behavior in compartments (deposition/condensation, radionuclide decay, and agglomeration)*.
- Assess sodium-chemical reactions in air-filled spaces (event specific).
- Assess barrier performance for licensing basis event with radiological consequences, and Design Basis Accidents (includes cliff edge effects, considerations for severe accidents, EPZ methodology, and others).

**phenomena also considered in the cover gas region*



Adapted from ANL-ART-49

Site Boundary



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Load Following w/ Integrated Energy (Thermal) Storage

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Load Following w/ Integrated Energy (Thermal) Storage

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Energy Island Capacity Optimization

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Energy Island –
Thermal Storage System

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Energy Island – Steam Generator Equipment

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Energy Island –
Turbine, Generator and Feedwater Systems

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Natrium Hybrid Main Control Room – Nuclear Island + Energy Island Human Machine Interfaces

[[

- Nuclear island control system (NIC) operations independent from energy island control system (EIC) operations
- Group view display system (GVDS)
- No safety-related (SR) action initiated from the main control room (MCR)
- Power runback (DL2) can be manually initiated or stopped by operators for operation flexibility
- Manual reactor trip or primary sodium pump trip can be initiated by an operator (for DID purpose only)
- Fuel handling control room located in the fuel handing building is independent of MCR
- All plant parameters (SR, or non-safety) available on NIC video display units for ease of operation.

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RPS Reactor Trip Parameters

Note: Current trip setpoints are under continued development and do not reflect the final selection of parameters and inputs for the plant design or licensing basis.

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Basic SCRAM Design Transient

Sequence of events following a SCRAM function:

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Basic SCRAM Design Transient (continued)

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Basic SCRAM Design Transient (continued)

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Basic Runback Design Transient

Sequence of events following a RUNBACK function:

[[

]]$^{(a)(4)}$

Basic Runback Design Transient (continued)

[[

]]^(a)(4)

Uncontrolled Rod Withdrawal

[[

]]$^{(a)(4)}$

Uncontrolled Rod Withdrawal

[[

]]$^{(a)(4)}$

Uncontrolled Rod Withdrawal

[[

]]$^{(a)(4)}$

Uncontrolled Rod Withdrawal

[[

]]$^{(a)(4)}$

Uncontrolled Rod Withdrawal

[[

]]$^{(a)(4)}$

Loss of Offsite Power

- Event sequence (long term response):

[[

]](a)(4)

Loss of Offsite Power

[[

]](a)(4)

Loss of Offsite Power

[[

]]$^{(a)(4)}$

Loss of Offsite Power

[[

]]^(a)(4)

Loss of Offsite Power

[[

]]$^{(a)(4)}$

Loss of Offsite Power

[[

]]$^{(a)(4)}$

Questions?

Acronym List

AHX – sodium-air heat exchanger
ANL – Argonne National Laboratory
BLTC – bottom loaded transfer cask
CCCS – core component conditioning station
CFR – *Code of Federal Regulations*
CRDL – control rod drive line
CRDM – control rod drive mechanism
DBA – design basis accident
DID – defense-in-depth
DL – defense line
ECI – export controlled information
EI – energy island
EIC – energy island control system
EPZ – emergency planning zone
ESS – energy island salt heat transport system
EVHM – ex-vessel handling machine
EVST – ex-vessel storage tank
FTP – fuel transfer port
GVDS – group view display system
HX – heat exchanger
IAC – intermediate air cooling system
IHT – intermediate heat transport system
ISP – intermediate sodium pump
IVTM – in-vessel transfer machine
LOCA – loss of coolant accident
MCR – main control room
NI – nuclear island
NIC – nuclear island control system
NQA – Nuclear Quality Assurance
NSS – nuclear island salt heat transport system
P/F – power to flow
PIE – post-irradiation examination
PRC – pin removal cell
PSP – primary sodium pump
RAC – reactor air cooling system
RPS – reactor protection system
RBFP – reactor building floor plug
RIS – reactor instrumentation system
RV – reactor vessel
SFR – sodium-cooled fast reactor
SR – safety-related
XIS – nuclear instrumentation system