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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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TERRESTRIAL ENERGY USA (TEUSA) SUBCOMMITTEE

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OPEN SESSION

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THURSDAY

MARCH 20, 2025

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The Subcommittee met via Video-Teleconference, at 8:30 a.m. EDT, Scott P. Palmtag, Chair, presiding.

COMMITTEE MEMBERS:

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ROBERT P. MARTIN, Member

DAVID A. PETTI, Member
THOMAS E. ROBERTS, Member
MATTHEW W. SUNSERI, Member

ACRS CONSULTANTS:
DENNIS BLEY
STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:
CHRISTOPHER BROWN

TABLE OF CONTENTS

P R O C E E D I N G S

8:30 a.m.

CHAIR PALMTAG: Good morning. This meeting will now come to order. This is a meeting of the Terrestrial Energy Subcommittee on the Advisory Committee on Reactor Safeguards. I am Scott Palmtag, chair of today's subcommittee meeting. ACRS members in attendance in person are Ron Ballinger, Matthew Sunseri, Greg Halnon, Craig Harrington, Robert Martin, Dave Petti, Tom Roberts, and myself. ACRS members in attendance virtually are Vesna Dimitrijevic and Vicki Bier. We have two consultants today virtually by Teams, and that's Steve Schultz and Dennis Bley. If I have missed anyone, either ACRS members or consultants, please speak up now.

Christopher Brown of the ACRS staff is the Designated Federal Officer for the meeting. No member conflicts of interest were identified for today's meetings. We have a quorum for today's meeting.

During today's meeting, the subcommittee will receive a briefing on the topical report and staff's draft safety evaluation for Terrestrial Energy principal design criteria (PDCs) for the integral molten salt reactor (IMSR) structure systems and components, Revision C. The PDCs are integral to the review of the unique aspects of the design. In addition, PDCs aid in the NRC staff's evaluation of applicable regulations and allow the NRC staff to assess with reasonable assurances that an advanced reactor technology will conform to the proposed design base with adequate margins of safety. We are reviewing this topical report because it serves as a foundation for the safety design approach for the IMSR.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, or FACA. The NRC implements FACA in accordance with our regulations. Per these regulations and the committee's bylaws, the ACRS speaks only through the published letter reports. All member comments should be regarded as only the individual opinion of that member, not a committee position.

All relevant information related to ACRS activities, such as letters, rules for meeting participation, and transcripts are located on the NRC public website and can easily be found by typing about us ACRS in the search field on NRC's homepage.

The ACRS, consistent with the agency's value on public transparency in regulation of nuclear facilities provides opportunity for public input and comment during our proceedings. We have received no written statements or requests to make an oral statement from the public. However, we have also set aside time at the end of this meeting for public comments.

Portions of this meeting may be closed to protect sensitive information, as required by FACA and the Government in the Sunshine Act. Attendance during the closed portion of the meeting will be limited to the NRC staff and its consultants, applicants, or licensees. And those individuals and organizations will have entered into an appropriate confidentiality agreement. We will confirm that only eligible individuals are in the closed portion of this meeting when it starts.

The ACRS will gather information, analyze relevant issues and facts, and formulate proposed conclusions and recommendations, as appropriate, for deliberation by the full committee. A transcript of this meeting is being kept and will be posted on our website.

When addressing the subcommittee, the participant should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. If you are not speaking, please mute your computer on Teams or by pressing *6 if you're on the phone.

Please do not use the Teams chat feature to conduct sidebar discussions related to the presentations. Rather, limit the meeting chat function to report IT problems.

For everyone in the room, please keep all your electronic devices in silent mode and mute your laptop microphone and speakers. In addition, please keep sidebar discussions in the room to a minimum because the ceiling microphones are live.

For the presenters, your table microphones are unidirectional and you'll need to speak into the front of the microphone straight-on to be heard. You also have to bring them fairly close to your mouth.

Finally, if you have any feedback for the ACRS about today's meeting, we encourage you to fill out the public meeting feedback form on the NRC's website.

We will now proceed with the meeting, and I'd like to call on John Segala, branch chief of NRR, for opening remarks.

MR. SEGALA: Thank you, chair and subcommittee. Good morning. I'm John Segala. I'm chief of the Advanced Reactor Licensing Branch 2 in the Division of Advanced Reactors in Non-Power Production and Utilization Facilities in NRR.

Over the past six years, Terrestrial Energy has been having pre-application engagement with the NRC staff on their integral molten salt reactor design. During that time, the NRC staff has reviewed and provided written feedback on eight white papers and one technical report. The NRC staff is also currently reviewing a topical report on postulated initiating events, as well as a technical report on source term. Both of these reports have been preceded by white papers where we have provided written feedback on them.

As you mentioned, we're here today to have Terrestrial and the NRC staff brief the ACRS subcommittee on Terrestrial's principal design criteria topical report and the NRC staff's safety evaluation. To help provide context for this topical report, Terrestrial is going to present a design overview of their IMSR, and that should help add a framework for the discussion today.

We understand that ACRS is maybe initially thinking that they would not issue a letter on this topical report. The NRC staff is not requesting one, but we do look forward to having discussions today with the subcommittee and hearing your input and your feedback on this topical report. I believe that completes my opening remarks.

Okay. Thank you. All right. We'll now turn it over to Darren Love, who will give a reactor design overview summary.

MR. AKSTULEWICZ: So, chairman, I'll be introducing folks at the table first to help you. To my far left is Simon Irish, the president of Terrestrial Energy. To his right is Darren Love, who is the director of engineering, and he will be doing the presentation on the systems and structures for the IMSR. My name is Frank Akstulewicz. I am the licensing manager for regulatory efforts for Terrestrial Energy here in the U.S., and to my right is Bill Smith who is the senior vice president for operations and engineering. Also in the audience is the vice president for business development and project management.

So I'd like to thank the committee for the opportunity to come before you and present information about the IMSR. This is our first opportunity to present to you on this particular model. Simon would like to have a few introductory remarks before we get into the presentation, so if that's okay.

MR. IRISH: Thank you, Frank. Chairman, it's a pleasure to be here today. We, as a company, have spent the last over a decade committed to developing the IMSR system. We have, over the last two years, recognized market demand is such that we are accelerating that development. This meeting today is an important milestone in our regulatory engagement with the NRC and represents our first topical report.

We have come here today, I think, with a full technical team and look forward to answering and assisting in the answers of any questions that may be discussed by the ACRS this morning. Thank you very much.

MR. LOVE: Good morning, chairman and fellow ACRS members. I'm grateful for the opportunity to present the design overview of the IMSR. My name is Darren Love. I am the engineering director for Terrestrial Energy. The engineering department of Terrestrial Energy is responsible for design, development, modeling, and simulation activities for the IMSR nuclear power plant. I've been with Terrestrial Energy for four years, serving first as a mechanical engineering manager prior to becoming the engineering director. Prior to joining Terrestrial Energy, I worked within the oil and gas industry performing fitness-for-service inspections and engineering analysis services for many of the world's leading refineries.

The integral molten salt reactor technology builds upon the extensive research conducted at Oakridge National Laboratories. The concept of the molten salt reactor dates back to the late 1940s with the Aircraft Reactor Experiment being the first molten salt reactor to operate from 1953 to 1954. The ORNL further advanced the MSR developments with the Molten Salt Reactor Experiment, which extensively operated from 1964 to 1969.

Following the MSRE, research continued on the Molten Salt Breeder Reactor and the Denatured Molten Salt Reactor. The DMSR introduced two innovations critical to the commercial viability of MSRs. First was the use of low-enriched uranium as the fuel source and a once-through fuel cycle enhancing proliferation resistance.

After a period of dormancy, interest in MSRs was renewed in the early 2000s as part of the Generation for Nuclear Reactor initiative. ONRL subsequently developed the SM-AHTR concept, which introduced the concept of the cartridge core design as a key innovation.

In 2012, Terrestrial Energy was founded to build upon these advancements, incorporating the use of standard assay LEU once-through fuel cycle and the integral core architecture into the IMSR core unit. The successful operation of the MSRE validated the feasibility of liquid-fueled MSRs. However, there are significant design differences between the MSRE and the IMSR, like the advancements in the fuel cycle strategy, safety, and commercial deployment considerations.

The IMSR plant is designed to provide high defense customized co-generation for industrial applications. It consists of two distinct facilities: the nuclear facility which houses the reactor and is subject to nuclear regulation and the thermal and electric facilities or a separate non-nuclear facility that converts thermal energy into industrial heat or electricity.

The nuclear facility labeled A on this slide follows a standardized design and operates independently from a thermal electric facility. In this dual IMSR configuration shown on this slide, it produces 884 megawatts of thermal energy at a supply temperature of 585 degrees Celsius.

MEMBER HALNON: Darren, this is Greg.

Before you get too far, what do you mean by standardized? I mean, this is a nonstandard design. Are you talking about the building layouts, the control room and that sort of thing, is standard or --

MR. LOVE: The standard IMSR design so --

MEMBER HALNON: Standard IMSR. Okay.

MR. LOVE: With the IMSR, the configuration or the site-specific configuration would occur in the thermal and electric facility, but the nuclear facility would be standard amongst all designs.

MEMBER HALNON: Keep in mind this is our first interaction with this. I know you've done a lot in Canada and whatnot, but we don't know what standard is yet.

MR. LOVE: Terrestrial Energy standardized design.

MEMBER HALNON: Okay. Thanks.

MR. LOVE: Thermal electric facility labeled as B is a non-nuclear installation that receives thermal energy from the two IMSR core units and can generate 822 megawatts of thermal input or 390 net of electrical power or a flexible combination of heat and electricity based on customer needs. Additionally, the IMSR nuclear power plant can incorporate molten salt, thermal energy storage and buffering, enhancing the load-following capability, and optimizing commercial performance. The end use of IMSR-generated power could be any industrial facility providing heat, electricity, or a combination of both.

This slide presents the major buildings of the IMSR plant. Within the nuclear facility, there are two primary structures: the reactor auxiliary building (RAB) labeled RAB 1 and RAB 2 and the common control building (CB). Each RAB houses a single IMSR core unit, along with its necessary nuclear and support systems to transfer heat from the reactor to the thermal electric facilities. These systems include the IMSR core unit, the heat transport systems, the process and emergency cooling systems. The control building is centrally located between the two RAB structures and contains the main control room for operating both core units in RAB 1 and RAB 2.

The thermal electric facility is located outside the nuclear facility perimeter. It consists of a turbine building for each of the operating core units and its corresponding RAB.

Based on commercial requirements, the turbine building and steam generation building may feature different heat transport configurations. For electrical generation applications, a salt-to-water steam generation system will produce super-heated steam to drive a non-nuclear industrial steam turbine and conventional power equipment. The turbine building also contains the feedwater, steam, and electrical systems for heat and power generation.

Located within each RAB or auxiliary building, there is a single operating IMSR core unit. The IMSR core unit is a graphite-moderated thermal spectrum and has a replaceable core unit on a seven-year replacement. It uses standard assay low-enriched uranium and is a liquid fuel molten salt reactor. It uses eutectic fluoride-based salt as both the fuel and the coolant.

MEMBER PETTI: Just a question. What's the power density?

MR. LOVE: Power density --

MEMBER PETTI: Of the reactor, the core --

MR. LOVE: So each core produces 442 megawatts thermal.

MEMBER PETTI: The power density could, could trigger --

MR. LOVE: Fair enough. I would say that we'll come back to you on that question.

MEMBER PETTI: I mean, light water

reactors sit at one number. I'm just trying to put it in the spectrum.

MR. LOVE: The IMSR core unit includes the integrated primary pumps, along with emergency heat removal systems within the IMSR unit. The IMSR design includes a passive negative temperature reactivity coefficient that passively controls the operation of the core unit.

Within the simplified flow diagram, on the left-hand side, it shows the IMSR core unit. Molten salt is circulated around the IMSR core unit through the integral pumping system through the graphite core to generate fission heat, which transfers the heat through the primary heat exchangers to a secondary coolant system, or SCS, which utilizes a non-nuclear fluoride-based secondary salt loop.

Heat is transferred from the SCS to the tertiary salt loop where site-specific design configuration allows for different options for power generation ranging from electrical generation for both on- and off-grid applications, process heat requirements, and grid stabilization programs such as backup power generation sources such as wind and solar.

I'll turn it over to --

MEMBER PETTI: Just another question. What's the pressure of the secondary loop?

MR. LOVE: The secondary loop is slightly higher than the primary loop.

MEMBER PETTI: In the atmospheric.

MR. LOVE: In the atmospheric.

MEMBER PETTI: Because we've had other designs come in the issue of a steam generator -- because I'm assuming that steam pressure is pretty high.

MR. LOVE: So in the secondary salt loop, the SCS is a fluoride-based salt, so going into the IMSR is fluoride salt, a near atmosphere --

MEMBER PETTI: I'm saying in the steam generator, right, that steam is pretty high pressure to feed the turbine.

MR. LOVE: Yes.

MEMBER PETTI: If there were steam generator tube break, high-pressure steam would come back into the secondary loop and, if it's not designed for high pressure, it could cause a failure. That's the safety issue.

MR. LOVE: Oh, you can't see my notes there, so we have the primary heat exchanger inside the IMSR core unit. We have a secondary heat exchanger which translates the secondary non-nuclear fluoride salt to a tertiary loop, and that tertiary loop then goes through -- there's a --

MEMBER PETTI: Okay. So it's the tertiary loop then.

MR. LOVE: The tertiary loop there, yes.

MEMBER PETTI: Okay.

MR. LOVE: So you have two heat exchangers between the water and the salt loop or the steam generation system. And the steam generation system outside the nuclear facility would be in the thermal electric facility and would be sufficiently protected from --

MEMBER ROBERTS: A similar question on this figure. This is Tom Roberts. The tertiary salt loop is not shown as feeding the steam generator. Is that also a tertiary salt loop that feeds the primary and the steam generator?

MR. LOVE: Yes. On this slide, it's trying to show different configurations, potential, so, yes, it would be the tertiary salt loop goes from the secondary heat exchanger to the steam generation heat exchanger in the case of pure electrical power generation.

MEMBER ROBERTS: So in Dave's question, it looks like the issue would be a steam generator tube leak which then feeds back to the secondary heat exchanger, which could feedback to the primary heat exchanger to something --

MEMBER PETTI: If they're not designed to handle the pressure. You get this propagation, right.

MR. LOVE: Yes, absolutely.

MEMBER PETTI: Before you move on, just another question. I don't know if we're going to move into something else. Obviously, I've got some questions on the design.

DR. BLEY: This is Dennis Bley. Along that same line, the secondary heat exchanger, I know this is just a cartoon, but what would appear to be the shell side that holds the salt for the tertiary loop, is that common to all three of those paths to the steam generator to the process heat and to the grid services?

MR. AKSTULEWICZ: This is Frank Akstulewicz. I'm not sure I understand your question. Could you try to explain it again or your question again?

DR. BLEY: From the secondary heat exchanger, there are three that goes out. One goes to the steam generator heat exchangers, one goes to what you call processing uses, and the third one goes to grid services. Is the salt that's moving through those three loops, it's common inside the secondary heat exchanger; is that right?

MR. AKSTULEWICZ: Sorry. The figure showing potential different configurations all at the same time, you would likely -- different applications are shown here, so it's more of a pictorial representation. You wouldn't have all --

DR. BLEY: Oh, well. We'll see the details later. Go ahead.

MR. AKSTULEWICZ: Okay. But what you're saying is is the configuration would be one of these three choices.

MR. LOVE: Yes.

DR. BLEY: Oh, okay. You wouldn't have a situation where you'd have multiple uses there. Okay.

MR. LOVE: Yes, exactly.

MEMBER PETTI: In terms of the core life, I assume it's the graphite that's limiting.

MR. LOVE: Correct.

MEMBER PETTI: Do you know what the damage -- I'm assuming it's sort of centerline, you know, in the center of the core where the damage is the greatest, at least that's what's causing the --

MR. SMITH: For the record, it's Bill Smith from Terrestrial Energy. Yes. The graphite is assumed to last seven years based on its turnaround and crossover point being seven years. We have actually --

MEMBER PETTI: So, yes, I know about graphite. So what's the BPA that you're reaching?

MR. SMITH: Above 21.

MEMBER PETTI: In seven years. Okay. And so your -- its limit crosses back from zero to zero. There's different criteria if you go back in time with graphite.

MR. SMITH: Yes, there are.

MEMBER PETTI: Okay.

MR. AKSTULEWICZ: Okay. Thank you, Darren. I'm going to take over the rest of the public presentation. My name is Frank Akstulewicz. My history has been a bachelor's degree in nuclear engineering from Penn State. I came to work for Bechtel Power Corporation in their Gaithersburg office back in the early 70s. I was involved in the construction of the Calvert Cliffs and Grand Gulf facilities.

From there, I moved on to the NRC and I've spent 42 years in service with the NRC in a number of positions across the board, including Senior Executive Service in which I was the deputy director or the director for new reactor licensing for ten years. I was in place when we were doing all the COL and design certification licenses under Part 52.

I spent two terms or, I guess, sessions I'll call them with two different commissioners as technical assistants. One was with Commissioner Jeff Merrifield, and I retired from Commissioner Annie Caputo's office in 2019. Since that time, I've been working with Terrestrial as their licensing manager facilitating their regulatory engagement activities.

So the next slide talks a little bit about what we have been doing. John mentioned it earlier, we're been actively engaged with the NRC with white papers, technical reports, and topicals. Some of them are listed there. John mentioned we had eight papers submitted. I went back and checked; that's actually right. It's nice to know.

We found the process of submitting white papers first before going to topicals or technicals to be very valuable. The engagement with the staff highlights a number of issues that we can facilitate or simplify to provide the review process being much simpler for them. So it's been very successful to this particular point. And as Simon mentioned earlier, this is the first topical report we have taken to the end. Hopefully, we'll get a final safety evaluation afterwards.

Our regulatory process for our risk-reduction activities in terms of licensing, we are currently engaged in pursuing a standard design approval under Part 52. That is the process we are engaged in. We believe it provides us the best capabilities to transition. If an opportunity for a construction permit would raise, we can convert the information we're developing directly into a CP application and move forward without any interruption going forward.

So next slide, please. So, again, back to the pre-application activities, well, first, I guess the most important thing there is the third bullet which highlighted the fact that we were one of the first designs to engage with the Canada Nuclear Safety Commission and a joint review. That was on our postulated initiating events technical report. We found that very successful in terms of highlighting the unique differences between our regulatory frameworks, but it also prepared us in terms of providing information that would be useful in actually converting the document to a topical report. John mentioned that we have that report in the moment. It does reflect lessons learned as an appendix to that topical report, and we're moving forward with that review.

A highlight that Simon didn't mention, but the IMSR has completed phase one and phase two of their vendor design review process with a positive outcome. So that, again, provided another measure of rigor with respect to understanding the design going forward and providing insights as to where additional work needs to be done.

We are engaged with IAEA. Our safeguards activities are using the IAEA state standards. We are also engaged with IAEA on a number of consultancy efforts going forward.

Okay. So getting into the development process for the topical report. One of the interesting things that we learned as part of this process is that fundamental safety functions in Canada and the U.S. are essentially the same. They may call them a little different, but, for all intents and purposes, the three primary ones that are listed there are identical between the two regulatory frameworks.

Canada does have a couple more that they add in, like their safety function of shielding and there's one for monitoring. Those are reflected in our classification of components activities. In terms of whether they will be safety related or not, we'll get into that discussion perhaps a little later this morning. But for all intents and purposes, that is the starting point for our PDC development.

As the chairman mentioned, our requirements for U.S. license applications under 52.137, specifically for standard design approval which is, again, our regulatory approach, and the Regulatory Guide 1.232 is the guidance for developing non-LWR principal design criteria that we used or followed for this particular process.

MEMBER PETTI: Just a question. You said that PDC are requirements for the U.S. Are they not requirements in Canada?

MR. AKSTULEWICZ: No, they're not.

MEMBER PETTI: Okay. Interesting.

MR. AKSTULEWICZ: Just as a quick note, but the Canadian exercise focuses on the safety functions. They don't have specific design criteria that you have to follow or develop.

CHAIR PALMTAG: This is Scott Palmtag. Since we're talking about the Canadian, can you describe exactly what that means, Canadian phase one and phase two?

MR. AKSTULEWICZ: Bill will take that one.

MR. SMITH: This is Bill Smith again. So Canadian Nuclear Safety Commission offers vendor design review phase one and phase two as a pre-licensing, so it's called the vendor design review. They have 19 criteria against which they measure the technology as to technology vendors, not licensees normally. And those criteria go from plant layout to decommissioning and everything in between. And then it's passed against the Canadian regulations, which don't include design criteria but safety functions, as Frank has said.

Phase one is a general sort of perspective do you have a clue and does your design have any chance of meeting the Canadian requirements, and then phase two just goes into deeper and deeper detail on each of the 19 criteria. They report back on it. We got their report in April of 2023, and the conclusion is no fundamental barriers to licensing so long as you continue to execute the design the way in which you said under a quality standard and you continue to do the R&D as you said and marry the two obviously; and if you continue that work, then this technology, as you've described it, is licensable. And there are some other elements to it.

CHAIR PALMTAG: So it's a pre-application stage.

MR. SMITH: Pre-application.

CHAIR PALMTAG: How would that relate to the NRC? It would start at the white paper stage.

MR. SMITH: Probably a little beyond white paper. To some extent, if I can assess it, topical technical reports -- sorry. A little beyond white paper, more topical technical reports.

The NRC and CNSC conducted a study of one of our processes, PIE, four years ago, so there was some coming together between Canadian and NRC on that particular topic. You know, just --

CHAIR PALMTAG: I'm just trying to understand how the --

MR. SMITH: Yes. So it's --

(Simultaneous speaking.)

CHAIR PALMTAG: -- Canadian stage versus --

MR. SMITH: -- more technical topical report as opposed to white paper. It's beyond white paper for sure.

CHAIR PALMTAG: So you said the Canadians are probably slightly ahead of the NRC in the reviews?

MR. SMITH: In the review of this particular technology, yes, but you --

CHAIR PALMTAG: Okay. Thank you.

MEMBER SUNSERI: So I think you touched on something, while we're talking about Canadians. I know the U.S. NRC has a memorandum of understanding with the Canadian CNS about accepting certain technical explanations or whatever. So have any of your topical reports been accepted by Canada and are part of the MOU with the U.S.?

MR. AKSTULEWICZ: So Canada does not review topical reports.

MEMBER SUNSERI: Okay.

MR. AKSTULEWICZ: That's not part of their process. They will review design information, they will review information associated -- they will do audits if they want, but, as a structure, there's not a process in place for topical reports.

MEMBER SUNSERI: But Bill was describing something that was part of it, though, right?

MR. AKSTULEWICZ: So my understanding, and Bill can correct me if I'm wrong, think of the vendor design review as a 50,000-foot level construction permit application where it highlights the details of the systems, the engineering, to the degree that they can. For example, an example I was thinking when the chairman was asking a question was the QA. They do a pretty extensive review of the QA system too their standard for quality assurance. It's not the same as Appendix B or NQA-1. They have their own CSA standard.

But they look at that to see if there is a weakness in their quality assurance oversight, so it's more than just a white paper. I mean, they look at the procedures and stuff, too, to make sure that it's implemented appropriately.

MEMBER SUNSERI: You're probably wondering why we're asking all these questions, at least I don't know why but -- why I'm asking them. If there's any possibility that something has been reviewed by the Canadians and may not come before us because it's already been accepted or something, that's kind of where my head is going on all this.

MR. AKSTULEWICZ: Yes. That will be the case with the PIE topical report where this would be the second time that the staff, maybe even the third time that the staff, we have seen that document through its iterations because it started out as a white paper, then it went to a topical report after the evolution of the review as part of the joint -- and there's a joint report, right. So both regulators put out their findings in terms of issues that they still identified as part of their review that needed to be resolved, so our topical report speaks to the findings within that joint report moving forward to show how we evolved our process to align with feedback from both regulators.

MEMBER SUNSERI: You used an acronym for that? What is that?

MR. AKSTULEWICZ: Oh, the postulated initiating events, PIE.

MR. SMITH: Bill Smith again just to clarify. That document sits inside our engineered system and has gone through three revisions now, thanks to both regulators and, of course, our own feedback. So the document sits there as part of the engineering basis, as well. And I can't imagine a thing that would have been, currently anyways, reviewed in Canada that would not ultimately then get reviewed by the NRC staff, as well.

MEMBER HARRINGTON: This is Craig. Just to be clear, on some of the reviews, there was this joint project. Are your efforts pursuing both Canadian and NRC approval, is that both ongoing?

MR. SMITH: The Canadian one is not ongoing. We stopped it at the end of vendor design review phase two. It would only start again if there was actually a license application in Canada, which is not likely in the near term. So, right now, the focus is to continue the stream in our --

MEMBER HARRINGTON: Okay. Thanks.

MR. AKSTULEWICZ: Those are all good questions, and feel free to ask. We're here to provide insight and answer your questions to help you understand the design, obviously, and what we've done and where we're going.

Okay. The next slide that is up is the general process that was used to build the principal design criteria document. As you are probably well aware, because you've probably seen other topical reports on principal design criteria, the starting point is Regulatory Guide 1.232 that contains several sets of general design criteria, including those that are not specific to technology, and then appendices to that regulatory guide are specific to certain types of technologies.

The key point here was you start at those starting points, and then you try to understand what the safety basis for each of those principal design criteria that are in the reg guide, what is the safety function they're trying to fulfill and why they're important. Then you start looking at the other reference sets to see how well they match up with the technology-specific requirements for SSCs that are unique to what would be the IMSR in our particular case and then look at how or if you would need to modify the specific principal design criteria that are in the regulatory guide to align with your specific technology.

After doing rather exhaustive review of the details and the general guidelines, it turns out that the sodium fast reactor reference set is the closest set for the IMSR. It's not aligned in all cases, but it served as a really good starting point. And, obviously, we had to take departures from that reference set to be specific to our technology and, obviously, we submitted those departures to the staff and had them reviewed as part of the draft safety evaluation, you know, finds them acceptable at the moment.

MEMBER PETTI: Just a quick question. I'm sure you must be aware of the ANS has a set of criteria.

MR. AKSTULEWICZ: Yes, I was on that

committee.

MEMBER PETTI: So, I mean, similar then if you line them up to what you ultimately have --

MR. AKSTULEWICZ: Yes. There are some unique differences. For example, the ANS standard, their containment is built around a functional containment design. We do not use a functional containment. We have a leak-tight, you know, metal containment, so that's obviously a variant right there off the bat, right.

But the most important thing, and this is one of the challenges that we had as part of that committee work, was they employ a SARRDL, and what is that? A specified acceptable radiological release limit. We did not employ that, and that's part of our discussion later this morning. But I can say that one of the reasons is that a SARRDL is calculated depending on what day of the week it is and whatever meteorological condition you have because it's the value that would reach a certain regulatory criteria if you got a release. Well, depending on how the wind blows and the rain or whatever, those numbers are different day to day. And the other thing is you won't know that you've exceeded it until you actually have a release that exceeded it.

So we chose not to go that way. We chose to monitor parameters that would tell us where we are in the process rather than waiting until the end to figure out whether we've over-exceeded our limit. And we'll get into that a little more in the proprietary session.

MEMBER ROBERTS: So if I follow up on that, the presentation mentioned LMP, licensing modernization. I assume you're not using any of the LMP concepts or --

MR. AKSTULEWICZ: Right now, the plan is to not use any of the LMP concepts. We are or will be risk informed. You have to use some PRA information when you're developing your postulated initiating events, but we do not believe that it's necessary to use the LMP methodology to reach a successful safety outcome, so that's kind of where we are.

MEMBER ROBERTS: Okay. Thank you.

MR. AKSTULEWICZ: Okay. Again, the process here, we submitted an initial white paper. We got feedback that we have turned into the topical report. The resolution of that feedback is in the topical report. We did take departures from the standard language, and we can go through those in a little more detail in the following session.

We could not adopt all of the language as it was written. The NRC did come out and do a regulatory audit of very specific topics, and I think we'll spend a lot more time on that this -- I keep saying this afternoon but later this morning.

And so, as a summary, bottom line is, of the 64 principal design criteria that are in the reference set, and that would be the sodium fast reactor reference set, we were able to adopt 31 of them without -- I'm sorry, 20-something of them. That's not the right number. No, that's not right. I think the number is 28 or 23 -- 26, right. And this was a stumbling point, so there's a little confusion here because we thought that we were not modifying a couple of them, but, during the final phases of our review with the staff, they said, well, you're changing this little word here, so that it's such a minor edit, but it is a change so it's a change. So the figures there aren't accurate.

So the staff's figures in the safety evaluation are accurate. We said 26, and I think it's, again, it's like 26 that were modified and then 10 that were not adopted in their entirety. And the 10 that were not adopted, I think, are important to understand, and we'll have that discussion in a later session.

MEMBER HALNON: This is Greg. Did you have to do any new ones any different?

MR. AKSTULEWICZ: So that's a great question. The answer is, no, we did not need to identify a new one. There was a lot of discussion about whether we needed one for graphite, and I'm sure we'll have more discussion of that this afternoon. We believe that the way we phrased our principal design criteria to focus on material limits and performance requirements simplifies that, so we don't need a unique principal design criteria just for graphite.

CHAIR PALMTAG: I just wanted to state for the public record that most of these modifications, as you mentioned, were very small. This literally changed sodium to molten salt, but there are quite a few that we're going to have questions on, and we'll have significant questions when we go into the closed session.

MR. AKSTULEWICZ: Correct, yes. Again, good point, chairman -- thank you -- is that a lot of the modifications weren't substantive. Another example is changing the language from primary coolant to a fuel cell boundary, which is the same but just different vernacular. So it's those types of simplifications that highlighted a lot of the changes where modifications were made.

I think that ends my presentation. So I'll take any more questions if folks have them.

CHAIR PALMTAG: Any questions? Any questions online? All right. Let's take a quick two-minute break and switch out the NRC.

(Whereupon, the above-entitled matter went off the record at 9:15 a.m. and resumed at 9:18 a.m.)

MR. ROCHE: Good morning, chairman and members. My name is Kevin Roche. I'm a project manager in the Advanced Reactor Licensing Branch 2. I'm very late in the game for this project, so there's a number of folks that I'll touch on later who really contributed greatly, made a greater contribution than I did to where we are today, including members of the staff. We have Matt Gordon and Ben Adams will be online and were not able to attend in person. They'll be doing, along with Hanh Phan, the majority of the presenting.

So I'll move on to this portion. I'll talk briefly about the chronology. We've had this topical for a bit of time, and, hopefully, this will be kind of a capstone and we can move forward, as was discussed in a number of other topicals and technical reports that Terrestrial has.

And now I'll turn it over to Hanh Phan who will kind of walk us through the open portion of the staff's presentation.

So as I mentioned, there's a number of folks who contributed to the review of this topical. As I mentioned, Matt Gordon and Ben Adams will be two of the principal reviewers and are both online to answer your questions. Ben Parks is also online to help out. Adrian Muniz, Michelle Vega Rodriguez, and Lucieann were all project managers or have all been project managers during this time, so they all contributed much more so than I did to the review.

Moving on, we initially received this topical in 2023, held on it, went back and forth with Terrestrial. They submitted two different subsequent revisions, and here we are today. We issued the draft around the 20th of February, and we're here in front of you all.

And with that, I'll turn it over to Hanh.

MR. PHAN: Thank you, Kevin. Good morning ladies and gentleman. My name is Hanh Phan, senior PIE -- and also the technical lead for the -- project. I spent almost 40 years in nuclear and PRA, half of those in the professional labs and nuclear power plants, the other half at the NRC.

In the next two slides, I will briefly outline the purpose of the TEUSA PDC topical report. The staff reviewed strategy, key regulations, and relevant guidance. The main purpose of the technical report is to establish criteria to support -- and future IMSR license applications while it will strengthen compliance with the regulatory requirements of 10 CFR Part 50 and 52 associated with the PDC.

The staff reviewed strategy ensuring compliance with the regulatory requirements. Two, assessing conformance with the staff guidance, specifically Reg Guide 1.232 guidance, with developing principal design criteria of non-light water reactors. Three, evaluating deviations from Reg Guide 1.232 on IMSR design features. And, four, assessing the applicability of Reg Guide 1.232 on IMSR design features.

Next slide, please. This slide identifies the regulation relevance to the PDC in the context with the provisions in 10 CFR Parts 50 and 52, applications for CP, OL, DC, COL, SDA, and MLs. They all must submit PDC for their proposed facilities. The specific regulation attached to the PDC are provided in 10 CFR 50.34, 52.47, 52.79, 52.127, and 52.157.

Since TEUSA intends to update the standard design approval for the IMSR core units. Therefore, this is subject to 10 CFR 52.157(a)(3)(I).

Additionally, 10 CFR Part 50 is also applicable to -- which specifies requirements for the scope and content of the PDC for non-LWRs.

Next slide, please. Reg Guide 1.232 provides guidance for non-standard designers, applicants, and licensees in developing PDC for non-LWR design as required by the regulation.

As mentioned in the topical report, TEUSA chose to use sodium fast reactors design criteria in Reg Guide 1.232, Appendix B -- design criteria with some modification. In the closed session discussion, the staff specifically removed them.

This slide also mentions the draft ANS 20.2-2023. Just to clarify that this ANS guidance is still undergoing endorsement as the basis for this TR evaluation.

With that, I will turn it to Ben for the IMSR PDC overview.

MEMBER SUNSERI: Just a quick question. So the applicant mentioned they had to take, for my words, an exception to some of the examples. I thought we were building a reg guide that was technology inclusive. I guess we missed molten salt reactors that have the reference to sodium or a sodium fast reactor standard. Am I about face or am I missing something here?

MR. SEGALA: Yes, this is John Segala from the NRC staff. I was just going to say that, yes, back when we developed Reg Guide 1.232, it was a large effort. We looked at developing the advanced reactor design criteria which were general, and then we developed high-temperature gas-cooled reactor design criteria and sodium-cooled fast reactor design criteria, but we did not, at that time, develop molten salt reactor criteria because, back then, there wasn't a whole lot of information on molten salt reactors in terms of our competence and being able to develop design criteria at that time. But as they mentioned, since then, ANS is working on that and looking at revising the reg guide to add some criteria of formal salt reactors, but we did not do that at the time.

Does that answer your question?

MEMBER SUNSERI: Yes, it does. Thank you.

MR. SEGALA: And I'm just going to add that the reg guide is, you know, not a requirement. It's one acceptable way of coming up with your principal design criteria, and it was technology inclusive but it was based on the certain designs at the time that we were looking at. So the reg guide talks about the designs that we considered when we developed those, but it was always anticipated as a developer uses the reg guide to help develop their principal design criteria that they would have to look at the unique aspects of their design and customize the PDCs to be appropriate for their unique design.

MEMBER SUNSERI: Okay. I mean, that's kind of what I was thinking. I mean, the appendices are just examples of how the criterion were applied, so they could have just applied the criterion and not have to take exception, right?

MR. SEGALA: To the extent it applies to their specific design, you know, because these were done based on the set of designs that we considered back when we developed the reg guide. And so, you know, you can only do so much based on what you know at the time, but there's a lot of different, even though you do it for the technology, there's a lot of nuances and uniqueness as to specific designs that they would have to customize that.

MEMBER SUNSERI: I'm not throwing stones with these guys. They were before you for ten years. I mean, that's well within the envelope of when the reg guide was developed.

MEMBER ROBERTS: Hey, Matt, I think I agree with your thought. We had similar questions when we reviewed the heat pipe microreactor PDC topical a few months ago, and then also there's no appendix in the reg guide for heat pipe reactor. And they based everything on the ARDCs, the high-level principles, and then they looked at the individual examples for guidance. But I think that's the same thing they did here, so the term exception to SFR criteria probably isn't this number, I'm thinking. The reality, they just had to find a different way to meet the advanced reactor design criteria, which is generically acknowledged and explicitly stated to apply this technology. Yes, it was a good point.

MEMBER SUNSERI: Yes, I'm not trying to belabor the point. We're all looking for inefficiencies in the regulatory process, and so, you know, I'm just trying to wrap my head around where they are.

DR. BLEY: This is Dennis. Just remembering back and supporting what John was saying, when they first brought that reg guide to us, I think one of the reasons is they picked the ones where they had substantial experience and expertise in-house to do as their first examples and told us at the time they intended to extend it but they wanted to get it out as fast as they could. I think that makes sense.

Since then, there's been an awful lot of work at NRC looking at other kinds of designs, including molten salt and development of the codes to support their reviews, too.

CHAIR PALMTAG: This is Scott Palmtag. I appreciate, John, that you're looking at Reg Guide 1.232 for molten salt. I'm having a little trouble that there seems to bifurcation between the ANS and the Terrestrial because one is going down a functional containment and one is going down a real containment, so I don't know how that's going to work. And even for new technologies, that might be an issue.

But one thing I want to mention is I'm kind of troubled by all these PDCs for the Terrestrial are proprietary. It would certainly make this a lot easier if these were PDC that we could apply to different technologies. Is it normal for PDC to be proprietary? I guess not because they're usually from Reg Guide 1.232, at least the ones I'm seeing.

MR. ROBERSON: I'm not sure how to answer that. I can say -- this is -- Roberson, I'm a branch chief in IJMU. If you recall the Westinghouse --for proprietary and even some of the PDCs we've seen for the high-fission gas reactors, IC100 for instance -- it's not uncommon.

CHAIR PALMTAG: I guess that's another reason to get, if we can get 1.232 updated for both the heat pipe reactors and the molten salt to kind of get ahead of this before everything becomes proprietary. Just a comment. Thanks.

MR. ADAMS: Good morning, everyone. I'm Ben Adams, technical reviewer with the NRC staff. I'm in NRR DANU Technical Branch No. 1. This slide is really just an overview of the PDCs that were chosen. If you've seen the reg guide, this looks familiar.

Terrestrial did provide a justification for every single design criteria they chose. We're not going to be going over every single one today because we'd be here for a few days, but we're going to highlight the substantial ones.

Next slide, please. Okay. And this slide is an overview of what's in the CT evaluation. I won't spend too much time on this slide either. It starts off with the regulations and the guidance that are relevant, which we just went over a couple of slides ago. And then it highlights some relevant design information, and then it goes into the PDC selection.

In the safety evaluation, we binned up the discussion based on PDCs with no changes from the reg guide and then ones with minimal terminology changes and then ones with substantial technical changes. And then it goes to the limitations and conditions, which there are a few of, and then the conclusions.

Next slide, please. I guess that's it for me.

MR. PHAN: Thank you, Ben. So in conclusion --

CHAIR PALMTAG: This is Scott Palmtag again. I appreciate we can't go through each one of these, but can you at least tell us the numbers of the ones that may be contentious?

MR. ADAMS: We have slides prepared for the ones that I think ACRS will be interested in. For example, I think we have a single slide that just summarizes --

CHAIR PALMTAG: Yes, I understand a lot of this is proprietary. But I'm just saying, for the public record, can we at least tell which ones that we're discussing with the main ones?

MR. PHAN: Yes. So based on the staff reviews, we identified 26 PDCs -- in the quantification, 20 of those with minor -- changes and 18 of those important --

So in the closed session, the staff would focus on the 18 that would be most important for your information. So, mostly, if you'd like to know which of those, they are PDC 5, 20 - 29, reacting with control systems; PDC 10 on reactor design; PDC 12 on suppression of reactor power -- PDC 19, control rooms; PDC 41 through 43, relevance to the containment atmosphere; and PDC 79, cover and off-gas inventory maintenance. Those will be discussed specific in the closed session.

MR. ADAMS: Okay. Thank you, Hanh.

CHAIR PALMTAG: So this is Scott. Did you take the PSAR from Abilene Christian University's project in -- to see and compare their PDCs to what we came up with, being that Abilene was really the first molten salt PSAR? So did you have any comparison there that you can talk about?

MR. PHAN: We did not, but, Matt, would you please respond to this question?

MR. GORDON: Hi. My name is Matthew Gordon. In response, I am not aware of any overlap with the Abilene Christian University PDCs and this review.

CHAIR PALMTAG: Because one of the purposes advertised for ACU was to support the commercialization of molten salt. It would surprise me if you didn't utilize that research in what you've already approved in your PSAR when you looked at this PDC.

So maybe that's just a comment. It's just surprising that you wouldn't have used that significant research there, plus it's already been approved by the agency and -- So I'll make some comparisons as we go forward.

MEMBER PETTI: And just another question. Are the limitations and conditions proprietary? Can you give us a sense in the open session of what the limitations were?

MR. PHAN: Because the language and mostly the, not the design criteria, the language in the PDC -- be more specific --

MR. ADAMS: Yes. This is Ben Adams. The limitations in their entirety are not proprietary, but there are some proprietary sentences in there.

MR. ROBERSON: This is Grant Roberson, branch chief from DANU. I do want to circle back to the observation made about ACU. If you can confirm my understanding of this, Ben, because you did work on --

does use a -- which has already been noted as one difference from other IMSR, and that would be in consideration -- correspondence of the PDCs. Would you agree with that, Ben?

MR. ADAMS: I would agree with that, yes.

MEMBER HALNON: But it's a graphite-moderated liquid fuel.

MR. ROBERSON: I understand. I'm just observing at least one difference with the --

MEMBER HALNON: I realize there would be some differences, but there was a lot of experience gained. I assumed you would have went through that.

MEMBER MARTIN: Regarding the proprietary nature of the PDCs, I can appreciate, at this stage in the review process, to hold back because of the potential for any of these PDCs to evolve. But there is not a reactor that it serves the public whose PDCs are not otherwise open because, of course, they're all in Appendix A. If I was a member of the public, I would want to know the criteria for which, you know, my neighborhood plant has been designed to. I think that is very important, and I have a hard stop with that.

But, again, I can understand there's some evolution in this process and that, you know, that final decision can be made much later. I would hope that, when it comes time to finalize the safety analysis report, that those PDCs are then open to the public so they can critique them. You know, not everyone is, you know, a nuclear engineer, but there are some. There are some very smart people out there that will care about this.

Anyway, I'll throw that out there.

MR. PHAN: Thank you, gentlemen. We take your feedback seriously. And in conclusion, to ensure that the -- PDC are properly developed and implemented, the staff has imposed four limitations and conditions. We're going to present them in the closed session. The staff finds TEUSA provided a sufficient set of PDCs with the IMSR design, subject to the L&Cs. The proposed PDCs established the design, application, construction, testing, and performance design criteria -- to provide reasonable assurance that IMSR could be operated without undue risk to the public. And based on our evaluation, the staff conclusion is -- report, revisions --- is suitable for use in the future IMSR licensing application.

So this marks the end of our presentation for the open session. In the upcoming closed session, we will go into more details on the PDCs that are considered important to -- with that, we will answer any additional questions you may have.

CHAIR PALMTAG: Any additional questions for Terrestrial or for the NRC in the open session from the ACRS members or consultants?

I think it's time to move on to public comments. We have not received any written comments for this meeting, but I would like to open it up for public comments. If anyone in the public would like to make a comment, please raise your hand and unmute your microphone when it comes time. I can see one. Spencer Toohill. Do you want to go ahead and unmute your microphone?

MS. TOOHILL: Yes. Hi, there. Good morning, everyone. Thank you all for the opportunity for the public to ask any questions or comments. My name is Spencer Toohill, and I am with the Breakthrough Institute. And I really just have, hopefully, a simple question. I'm just interested in learning a bit more about what the next steps are here. Obviously, this will transition to the closed session for you all to discuss --

CHAIR PALMTAG: I'm sorry, Spencer, to cut you off, but this is for public comments only, not questions. If you do have any questions, I'd like to refer you to the Designated Federal Officer, Chris Brown, and his email can be found on the meeting notice.

MS. TOOHILL: Okay. Thank you for the redirect. I appreciate it. Thanks so much.

CHAIR PALMTAG: Do you have a comment?

MS. TOOHILL: No, I just had a question, so I'm all set. Thank you so much.

CHAIR PALMTAG: All right. Thank you. Was there another comment? Okay. Seeing none, I think we're going to go ahead and we'll close the open session and we'll move to a closed session. All right. Thank you.

(Whereupon, the above-entitled matter went off the record at 9:44 a.m.)

TERRESTRIAL ENERGY

Delivering carbon-free thermal and electrical energy

Development of Principal Design Criteria for the Integral Molten Salt Reactor (IMSR)
ACRS Subcommittee Meeting – Open Session
March 20, 2025

Introductions

Simon Irish, CEO and Director, Terrestrial Energy

William Smith P.Eng, Senior Vice President of Operations and Engineering, Terrestrial Energy

Francis Akstulewicz, Licensing Manager, Terrestrial Energy

Darren Love P.Eng, Engineering Director, Terrestrial Energy

Agenda

I	**IMSR Technology Overview**
II	Licensing Strategy
III	Preapplication Activities
IV	Overview of Topical Report Development Process
V	Conclusions

IMSR is based on MSR technology demonstrated at Oak Ridge National Laboratory (ORNL)

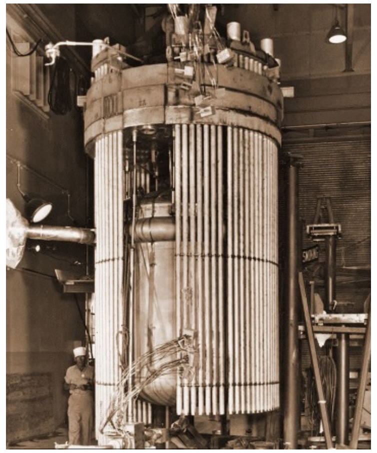
Based closely on molten salt technology demonstrated at ORNL. IMSR is a molten salt reactor that uses:

- Fluoride chemistry
- Under 5% LEU once-through fuel cycle
- Thermal spectrum
- Graphite moderator
- Integral core architecture

1958 -1969

First Molten Salt Reactor (MSR) research program started in the 1950s[1]

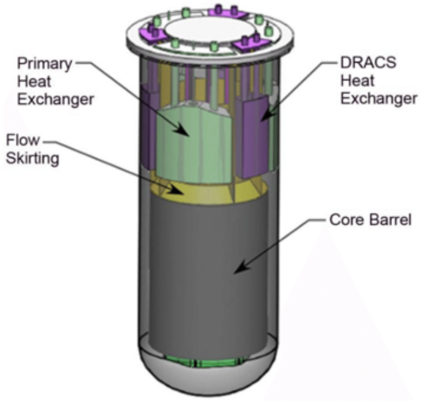
Molten Salt Reactor Experiment (MSRE) at ORNL highly successful and lays foundation for future molten salt reactor designs



2010

Small Modular Advanced High-Temperature Reactor (Sm-AHTR) design, using solid fuel and molten salt cooling[3]

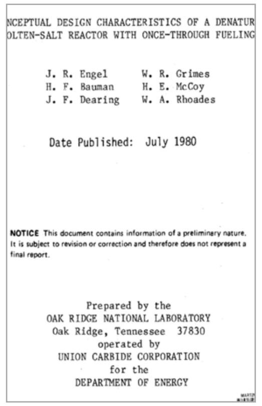
Key innovation: Cartridge core design



1980

Denatured Molten Salt Reactor (DMSR)[2] conceptual design developed at ORNL

Key innovation: Use of Low Enriched Uranium (LEU) with a once-through fuel cycle for strong proliferation defenses


NCEPTUAL DESIGN CHARACTERISTICS OF A DENATUR
OLTEN-SALT REACTOR WITH ONCE-THROUGH FUELING

J. R. Engel W. R. Grimes
H. F. Bauman H. E. McCoy
J. F. Dearing W. A. Rhoades

Date Published: July 1980

NOTICE This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

>2012

Terrestrial Energy's IMSR combines these critical innovations

Use of SA-LEU fuel with a once-through fuel cycle

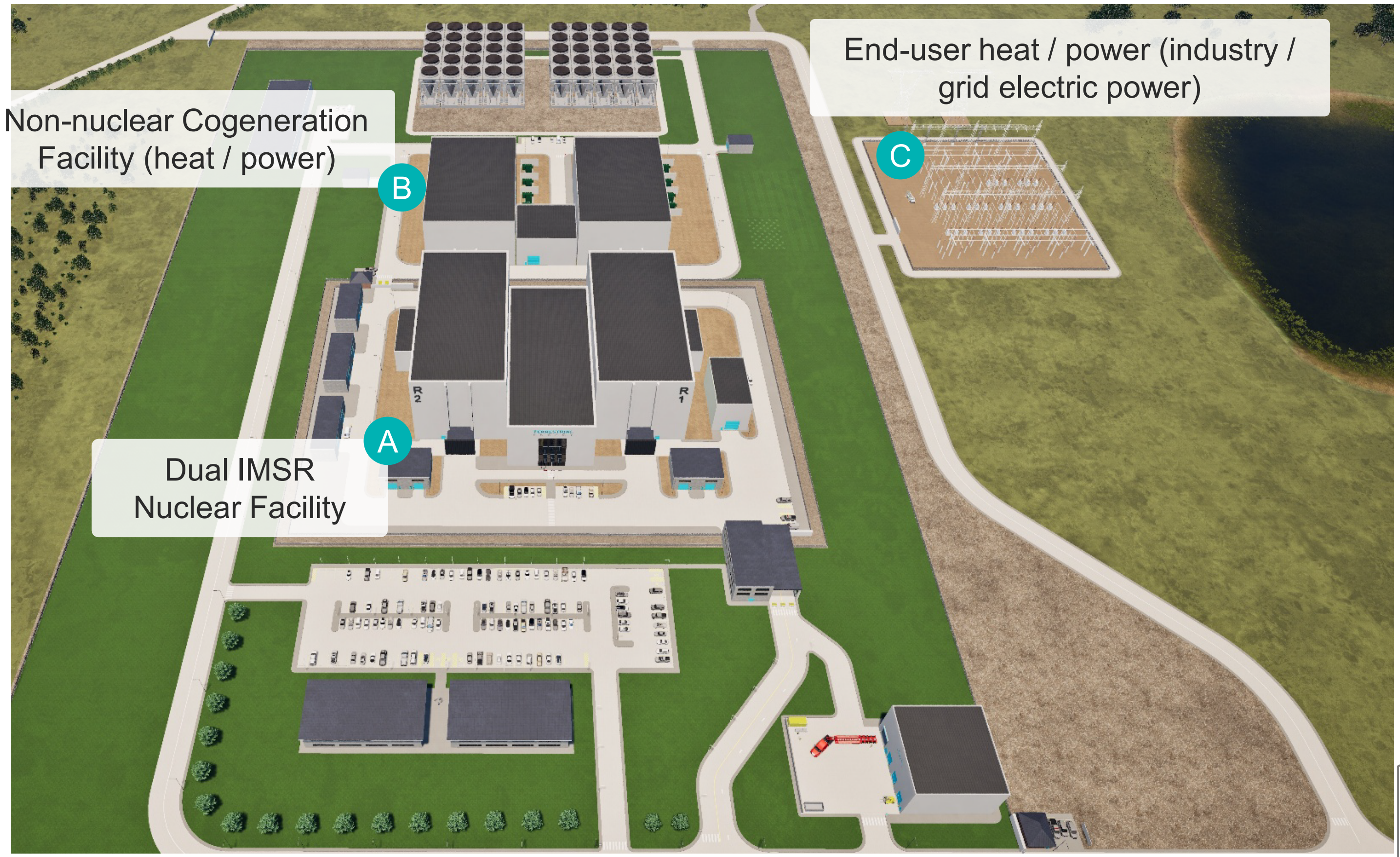
Integral core architecture

1. ORNL, Molten Salt Reactor History and ORNL-2474 Quarterly Progress Reports 1958-1976
2. ORNL, Conceptual Design Characteristics of a Denatured Molten-Salt Reactor with Once-Through Fueling
3. ORNL, Pre-Conceptual Design of a Fluoride-Salt-Cooled Small Modular Advanced High-Temperature Reactor (SmAHTR)

Source: ResearchGate; ORNL; Company

IMSR Plant is designed to deliver – “behind the fence” – customized cogeneration to industry

Separation of nuclear from thermal and electrical systems allows a standardized reactor design, while giving the end-user **the flexibility to use thermal, electric, or both**



A Standardized dual IMSR Nuclear Facility

- Subject to nuclear regulation
- Standardized, simplifying design and saving costs
- 884 MW (gross) thermal energy production for 585°C supply

B Customized non-nuclear Thermal and Electrical facility

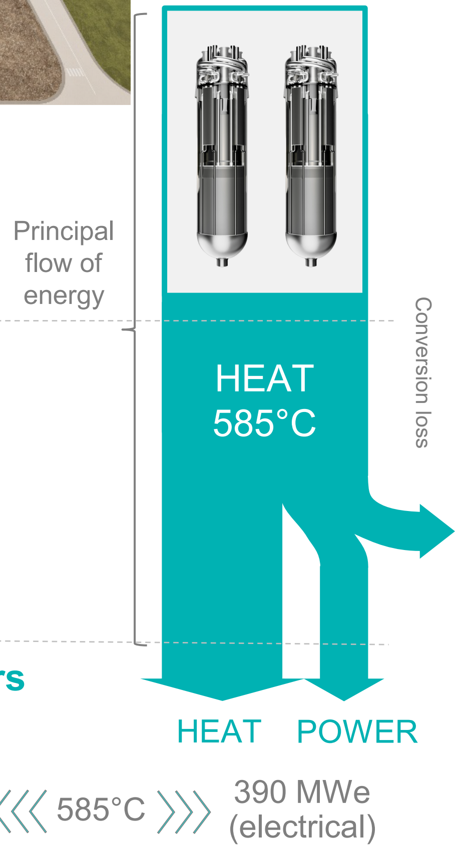
- Converts 884 MW (gross) thermal energy from two IMSRs to 585°C 822 MW (net) thermal or 390 MW (net) electric power for commercial supply – or any heat/electric power mix in between
- Can include molten-salt thermal energy storage and buffering to enhance its inherent strong load-following capability for commercial advantage
- Separate Nuclear Facility & non-nuclear Cogeneration Facility

C Prospective industrial cogeneration off-takers

- Chemical and petrochemical plant
- Hydrogen / ammonia / fertilizer plant
- Other industrials requiring clean heat & power

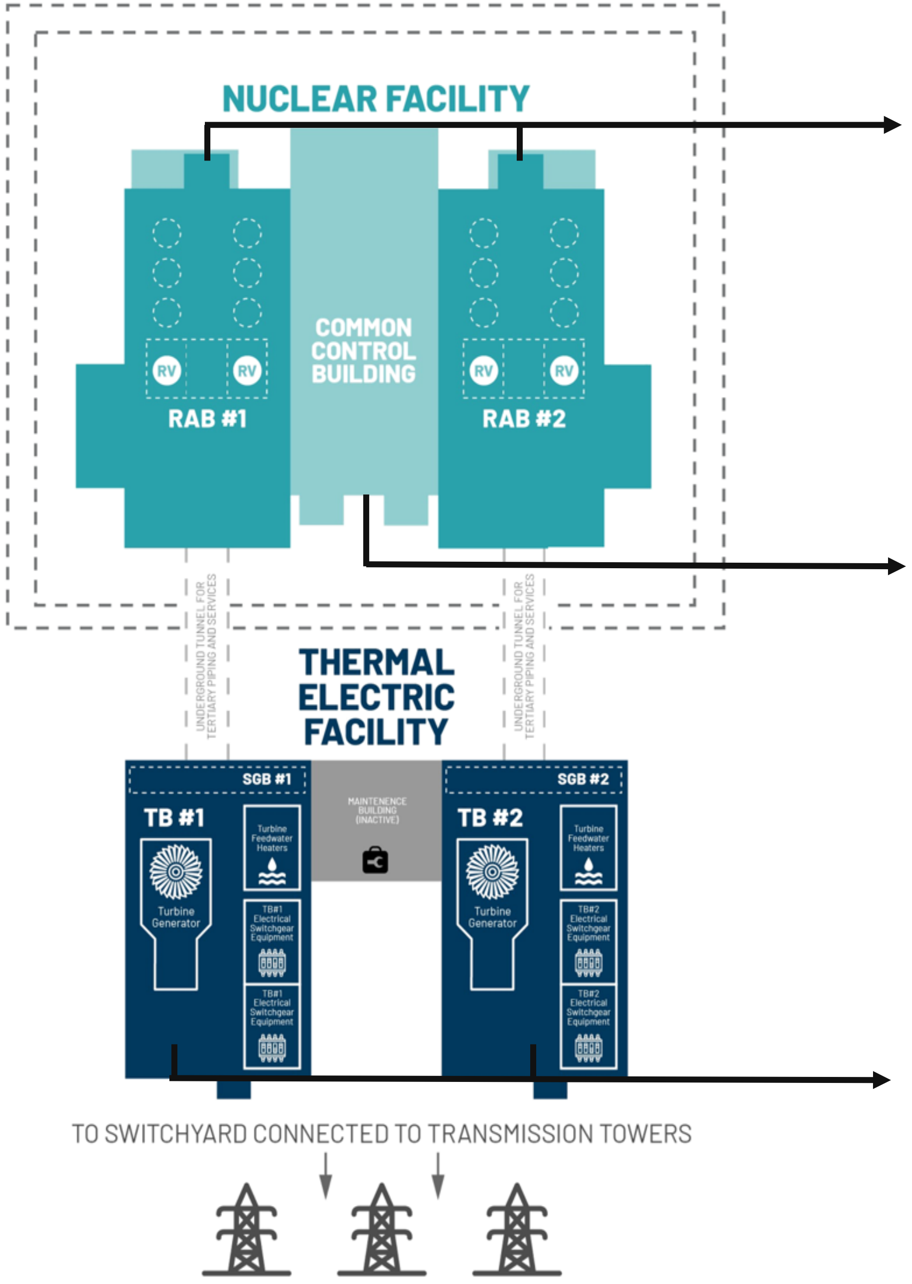
Prospective municipal off-takers

- Electric grid
- Desalination



Note: Example is for a dual reactor core IMSR Plant. Scaling up is possible.

IMSR Plant Layout



RAB Buildings	• Reactor Auxiliary Buildings (RAB), each containing an operating IMSR Core-unit and associated nuclear and support systems necessary to transfer heat in the reactor to the associated Thermal Electricity Facility.
Common Control Building	• Located between the two RAB structures, supports and provides services to both RAB units. Utilizes a common Main Control Room (MCR) for both RAB.
Turbine Buildings	• Each Turbine Building (TB) contains non-nuclear-grade, industry standard power equipment. The TB houses the Turbine Generator Set (TG), Condenser, and the associated feedwater, steam systems, electrical systems and other required equipment

IMSR Technology Overview

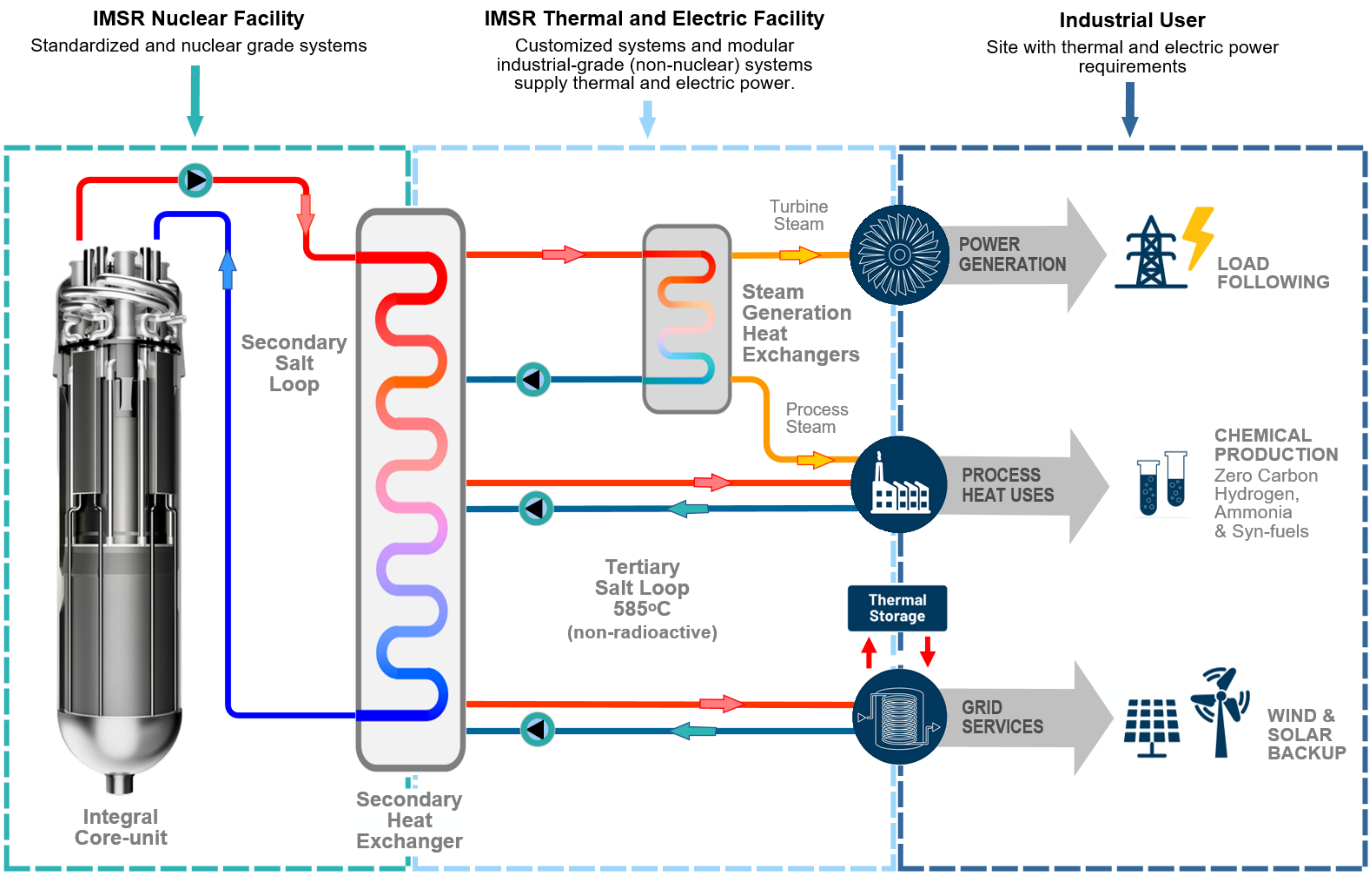
Graphite Moderated, Thermal Spectrum with Replaceable Core-unit (seven-year cycle)

Standard Assay Low Enriched Uranium <5% Enrichment and On-line Fueling

Liquid-Fueled Molten Salt Reactor. The molten salt act as both the Fuel and Coolant

Integrated Primary Pumps and Heat Exchanger with Emergency Heat Removal

Passive Reactivity Control (negative temperature reactivity coefficient)



Agenda

Licensing Strategy

Engage NRC with Preapplication Activities

- Regulatory Engagement Plan submitted
- White papers
  - Definition of IMSR Core-unit
  - Exemptions required under Part 52
  - Postulated Initiating Event Methodology (joint CNSC/NRC review)
- Technical reports
  - Modeling and Simulation of Off-Gas Source Term
- Topical reports
  - Principal Design Criteria for IMSR
  - Postulated Initiating Events Methodology

Pursue Standard Design Approval under Part 52

- Reduces regulatory risk
- Provides options for conversion to Construction Permit under Part 50

Agenda

Preapplication Activities

US NRC

- IMSR pre-licensing activities with the US - NRC commenced in 2018 with grant support from the U.S. Department of Energy (DOE)
- Several white papers and topical reports have been submitted to and reviewed by the US NRC as part of the licensing engagement plan in the US, including PDC TR
- IMSR was selected by the US NRC and the CNSC for the first cross-border joint review of a high temperature reactor technology

Canadian Nuclear Safety Commission (CNSC)

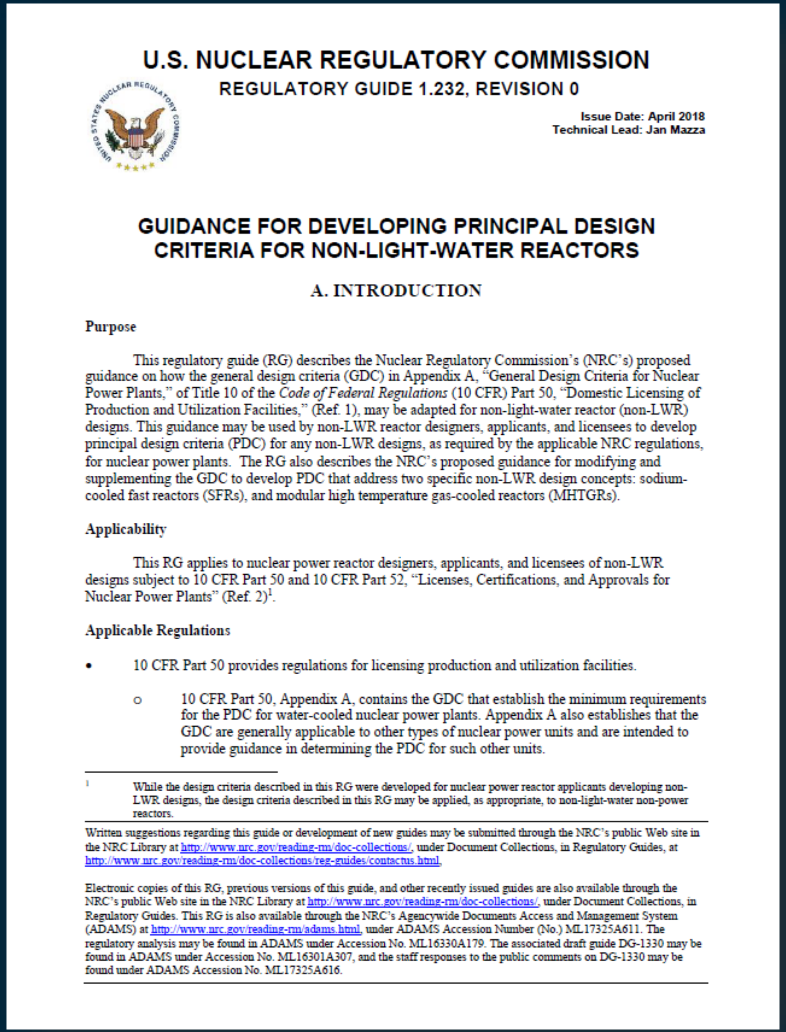
- completed Phase 1 and Phase 2 of the Vendor Design Review (VDR) of IMSR with positive conclusion

IAEA

- safeguards-by-design will facilitate future licensing application for the IMSR
- Significant interactions with the IAEA and the CNSC on safeguards-by-design (SBD)
- Valuable feedback from the IAEA is being considered in the detailed design phase

Agenda

PDC Development Process



U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.232, REVISION 0

Issue Date: April 2018
Technical Lead: Jan Mazza

GUIDANCE FOR DEVELOPING PRINCIPAL DESIGN CRITERIA FOR NON-LIGHT-WATER REACTORS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes the Nuclear Regulatory Commission's (NRC's) proposed guidance on how the general design criteria (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," (Ref. 1), may be adapted for non-light-water reactor (non-LWR) designs. This guidance may be used by non-LWR reactor designers, applicants, and licensees to develop principal design criteria (PDC) for any non-LWR designs, as required by the applicable NRC regulations, for nuclear power plants. The RG also describes the NRC's proposed guidance for modifying and supplementing the GDC to develop PDC that address two specific non-LWR design concepts: sodium-cooled fast reactors (SFRs), and modular high temperature gas-cooled reactors (MHTGRs).

Applicability

This RG applies to nuclear power reactor designers, applicants, and licensees of non-LWR designs subject to 10 CFR Part 50 and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2)[1].

Applicable Regulations

- 10 CFR Part 50 provides regulations for licensing production and utilization facilities.
  - 10 CFR Part 50, Appendix A, contains the GDC that establish the minimum requirements for the PDC for water-cooled nuclear power plants. Appendix A also establishes that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the PDC for such other units.

[1] While the design criteria described in this RG were developed for nuclear power reactor applicants developing non-LWR designs, the design criteria described in this RG may be applied, as appropriate, to non-light-water non-power reactors.

Written suggestions regarding this guide or development of new guides may be submitted through the NRC's public Web site in the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/, under Document Collections, in Regulatory Guides, at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html.

Electronic copies of this RG, previous versions of this guide, and other recently issued guides are also available through the NRC's public Web site in the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/, under Document Collections, in Regulatory Guides. This RG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under ADAMS Accession Number (No.) ML17325A611. The regulatory analysis may be found in ADAMS under Accession No. ML16330A179. The associated draft guide DG-1330 may be found in ADAMS under Accession No. ML16301A307, and the staff responses to the public comments on DG-1330 may be found under ADAMS Accession No. ML17325A616.

Fundamental safety functions in Canada and US are the same

- Control reactivity
- Remove heat from reactor and stored fuel
- Confine radioactive releases so regulatory criteria are not exceeded

PDC establish programmatic elements of a license that assure that the fundamental safety functions will be performed

PDC are requirements for US license applications

RG 1.232 establishes guidance for developing non-LWR PDC

Process for Selecting Initial Set of PDC

Non-LWR Crosswalk Table

The following table (Table 1) provides a summary and crosswalk between the LWR GDC contained in 10 CFR Part 50 Appendix A and the NRC staff's determination of their applicability to the ARDC, SFR-DC, and MHTGR-DC. For each design criterion, the table denotes the status (same as GDC, same as ARDC, modified for ARDC, modified for SFR-DC, or modified for MHTGR-DC). Table 1 also uses redline-strikeout to identify the design criteria titles that have been modified for non-LWRs. Words removed from the title are in red with a strikethrough and words that have been added are in blue and underlined. The actual ARDC, SFR-DC, and MHTGR-DC and NRC staff technology-specific rationale for adaptions to the GDCs are contained in Appendices A—C to this RG.

The table consists of five columns:

Column 1—Criterion Number
Column 2—Current GDC Title (from 10 CFR Part 50, Appendix A)
Column 3—ARDC Title/Status (showing conformity to or deviation from 10 CFR Part 50, Appendix A)
Column 4—SFR-DC Title/Status (showing conformity to or deviation from 10 CFR Part 50, Appendix A)
Column 5—MHTGR-DC Title/Status (showing conformity to or deviation from 10 CFR Part 50, Appendix A)

The table is divided into seven sections similar to those in 10 CFR Part 50, Appendix A:

Section I—Overall Requirements (Criteria 1–5)
Section II—Multiple Barriers (Criteria 10–19)
Section III—Reactivity Control (Criteria 20–29)
Section IV—Fluid Systems (Criteria 30–46) for ARDCs, and SFR-DC
Section IV —Heat Transport Systems (Criteria 30-46) for MHTGR-DC
Section V—Reactor Containment (Criteria 50–57)
Section VI—Fuel and Radioactivity Control (Criteria 60–64)
Section VII—Additional SFR-DC (Criteria 70–77) and Additional MHTGR-DC (Criteria 70–72)

RG 1.232, Rev. 0, Page 12

RG 1.232 contains several sets of general PDC that serve as starting points for technology specific PDC

Understand the safety basis supporting the general PDC before selecting the initial starting set of PDC

Understand the safety philosophy of the systems, structures and components in the reference PDC set to determine if the specific technology has similar or identical requirements

Perform a line-by-line examination of the selected reference set language to the SSCs of the specific technology and the expected safety functions that are to be performed

Sodium fast reactor reference set was closest to IMSR technology but not aligned in all cases

Departures from reference set criteria must be justified in a licensing basis topical report and be approved by the US regulator

Overview of PDC review and results

TE white paper initially submitted to NRC to begin regulatory engagement on PDC development process.

NRC feedback incorporated into topical report and documentation bases.

Departures from RG 1.232 reference set of design criteria are justified and approved by US regulator.

Not all reference set PDC are adopted without modification to make the criteria technology specific.

NRC performed regulatory audit to support their regulatory findings.

Of the 64 PDCs set by the reference RG 1.232:

- 31 were adopted without modification
- 23 were adopted with modifications to reflect IMSR specific design
- 10 were not adopted as not necessary for the IMSR

Agenda

NRC Staff Review of the Terrestrial Energy USA, Inc. Principal Design Criteria Topical Report for the Integral Molten Salt Reactor

ACRS Subcommittee Meeting (Open Session)
March 20, 2025

Kevin Roche, Project Manager
Hanh Phan, Senior Reliability and Risk Analyst
Ben Adams, Nuclear Systems Engineer
Matthew Gordon, Materials Engineer

Office of Nuclear Reactor Regulation (NRR)
Division of Advanced Reactors and
Non-Power Production and Utilization Facilities (DANU)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Agenda

- Review chronology
- Topical report (TR) purpose and review strategy
- Safety evaluation (SE) overview
- Conclusions

NRC Review Team

- Matthew Gordon, Materials Engineer, NRR/DANU (Technical Lead)
- Christopher Adams, General Engineer, NRR/DANU
- Joseph Ashcraft, Former NRC Staff, NRR
- Hanh Phan, Senior Reliability and Risk Analyst, NRR/DANU (IMSR Project Lead)
- Benjamin Parks, Senior Technical Advisor, NRR/DANU
- Kevin Roche, Project Manager, NRR/DANU (IMSR Project Manager)
- Adrian Muniz, Senior Project Manager, NRR/DANU
- Michelle Vega Rodriguez, Project Manager, NRR/DANU
- Lucieann Vechioli Feliciano, Project Manager, NRR/DANU

Review Chronology

- Jun 8, 2020: White Paper containing proposed Principal Design Criteria (PDC) for Integral Molten Salt Reactor (IMSR) submitted (ML20178A457)
- Aug 20, 2020: NRC staff provided comments (ML20304A561)
- Jan 17, 2023: TEUSA IMSR PDC TR, Revision 0 submitted (ML23025A066)
- Feb 17, 2023: TR accepted for review
- Sep 28, 2023: Closed clarification meeting
- Dec 29, 2023: TEUSA IMSR PDC TR, Revision B submitted (ML24053A168)
- Apr 8, 2024: PDC TR audit commenced
- Jul 2, 2024: Audit exited
- Jul 19, 2024: TEUSA IMSR PDC TR, Revision C submitted (ML24204A092)
- Aug 28, 2024: Closed clarification meeting
- Oct 7, 2024: Audit report issued (ML24233A246)
- Nov 4, 2024: Closed clarification meeting
- Feb 20, 2024: NRC staff's draft safety evaluation report issued (ML24339A121)

TR Purpose and Review Strategy

- Purpose of TR
  - Establish the PDC to support the design/future license applications referencing the IMSR
  - Demonstrate compliance with the relevant regulatory requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Parts 50 and 52 associated with PDC
- Review strategy
  - Ensure compliance with regulatory requirements
  - Review conformance with Regulatory Guide (RG) 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors"
  - Evaluate deviations from RG 1.232 in consideration of the key IMSR design features
  - Assess applicability of RG 1.232 appendices and guidance to novel IMSR design features

Regulations

- In accordance with the provisions of 10 CFR Parts 50 and 52, applicants for a construction permit (CP), operating license (OL), standard design certification (DC), combined license (COL), standard design approval (SDA), or manufacturing license (ML) must submit PDC for the proposed facility. Specifically, the following regulations pertain to the PDC:
  - 10 CFR 50.34(a)(3)(i), which requires, in part, that applications for a CP include PDC for the facility. An OL would reference a CP, which would include PDC
  - 10 CFR 52.47(a)(3)(i), which requires, in part, that applications for a standard Design Criteria (DC) include PDC for the facility
  - 10 CFR 52.79(a)(4)(i), which requires, in part, that applications for a COL include PDC for the facility
  - 10 CFR 52.137(a)(3)(i), which requires, in part, that applications for an SDA include PDC for the facility
  - 10 CFR 52.157(a), which requires, in part, that applications for a ML include PDC for the reactor to be manufactured
- 10 CFR Part 50, Appendix A provides requirements on the scope and content of PDC for non-light water reactors (non-LWRs):
  - "The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public."

Guidance

- RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors” (ML17325A611)
  - Appendices provide example advanced reactor design criteria
- Draft American National Standards Institute (ANSI)/American Nuclear Society (ANS) ANSI/ANS 20.2-2023, “Nuclear Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten Salt Reactor Nuclear Power Plants”
  - The NRC staff is in the process of reviewing ANSI/ANS 20.2-2023 for requested endorsement

IMSR PDC Overview

- TEUSA established the IMSR PDC as follows:
  - Section I – Overall Requirements (DC 1-5)
  - Section II – Multiple Barriers (DC 10-19)
  - Section III – Reactivity Control (DC 20-29)
  - Section IV – Heat Transport Systems (DC 30-46)
  - Section V – Reactor Containment (DC 50-57)
  - Section VI – Fuel and Radioactivity Control (DC 60-64)
  - Section VII – Additional (DC 70-79)

Safety Evaluation Overview

- Regulations and guidance
- IMSR design features (informational)
- IMSR PDC
  - PDC with no changes to the design criteria in RG 1.232
  - PDC with minor terminology changes
  - PDC with substantive technical changes
- Limitations and conditions
- Conclusions

Conclusions

- The NRC staff established four Limitations and Conditions (L&Cs)
- TEUSA provided a sufficient set of PDC for the IMSR design, subject to the L&Cs
- The PDC (subject to the L&Cs) establish the necessary design, fabrication, construction, testing, and performance DC for safety significant SSCs to provide reasonable assurance that the IMSR reactor could be operated without undue risk to the health and safety of the public
- The TEUSA PDC TR, Revision C, is suitable for reference in future licensing applications for the IMSR

Abbreviations

ANSI/ANS – American National Standards Institute/American Nuclear Society

CFR – *Code of Federal Regulations*

COL – Combined license

CP – Construction permit

DANU – Division of Advanced Reactors and Non-power Production Facilities

DC – Design criteria

IMSR – Integrated Molten Salt Reactor

L&C – Limitation and condition

LWR – Light water reactor

ML – Manufacturing license

NRR – Office of Nuclear Reactor Regulation

NRC – Nuclear Regulatory Commission

OL – Operating license

PDC – Principal design criteria

RG – Regulatory guide

SDA – Standard design approval

SSC – Structure, system, or component

SE – Safety evaluation

TEUSA – Terrestrial USA, Inc.

TR – Topical report