Lead-Cooled Fast Reactor (LFR)
Prof. Craig Smith, US Naval Graduate School, USA

Berta Oates: Okay, well, if we don't have any last-minute questions, I suggest that we just get started. By way of housekeeping, any questions can be posted in the Q&A pod. We will take those as time allows at the end. The presentation slide deck is available for download in the Files pod by clicking on that title. It should download a PDF copy right to your desktop. It will also be available at the end, plus the recording on the GIF website. As always, we have a survey online. The link to that is posted in the Notes pod.

And without any further ado, we will go ahead and get started. Doing today's introduction is Dr. Patricia Paviet. Patricia is with the Department of Energy. She is also the Chair of the GIF Education and Training Task Force. Patricia?

Patricia Paviet: Yes, thank you so much Berta. Good morning, everybody. A pleasure to have Prof. Craig Smith with us today. He is a Researcher, Professor of Physics, at the Naval Postgraduate School, Monterey, California, and he received his PhD in Nuclear Science and Engineering from the University of California, Los Angeles in 1975.

He is a Fellow of the American Nuclear Society and the American Association for the Advancement of Science. He has research experience in nuclear energy, radiation detection, and nuclear forensics. His previous employers include the US Army, Science Applications International Corporation, Booz Allen and Hamilton, and Lawrence Livermore National Laboratory, where he was a Deputy Associate Director and he led the Fission Energy Program.

Beginning in 2004, he became the Lawrence Livermore National Laboratory Chair Professor at the Naval Postgraduate School in Monterey. He serves as the US Observer Representative to the GIF Provisional System Steering Committee for the Lead-cooled Fast Reactor.

And without any delay, I give you the floor, Craig. Thank you again for volunteering to give this webinar.

Craig Smith: Well, thank you, Patricia, for the kind introduction, and welcome, everyone, to today's webinar on the lead-cooled fast reactor. Let me start by thanking the organizers of the webinar series and acknowledging the hard work and active involvement of the sponsors in making them a reality.

At the outset, I'll say a few words about the physics and motivation for fast reactors in general and then later discuss the specifics of the lead-cooled fast reactor.

At present, nuclear reactors produce more than 10% of the world's electricity and much higher levels than that in several countries, for example, the level in France is 72%, Belgium 50%, in Korea 30%, in the United States 20%, and similar levels in many other countries. However, current thermal reactors use only about six-tenths of the energy value in the mined natural uranium. They produce long-life transuranics as nuclear waste or spent fuel, and they operate with a relatively low level of efficiency in conversion of their released energy into electricity with typical power conversion efficiencies of about 33%.

Gen IV fast reactors, and the lead-cooled fast reactor in particular, can offer strong improvements to address these and other issues associated with the current generation of reactors.

Today's presentation provides some background on fast reactors and then a more detailed description of the development and current status of the LFR.

So this slide shows the agenda for this webinar, and I'll start with a recap of the basic physics of fast reactors, recognizing that this may be a refresher for some, but for others it may represent an important background. Next, I'll discuss the characteristics and challenges of advances LFRs and the historical development of the lead-cooled fast reactor within the Generation IV and Generation IV International Forum, or GIF, context. Following that, I'll describe the GIF reference system reactors and several additional LFR designs that are currently being developed. I'll wrap up with some summary comments and conclusions.

So first of all, a recap of fast reactor physics and their implications.

This graphic, courtesy of Dr. Bob Hill, who presented an earlier webinar on the sodium fast reactor, compares the neutron energy spectrum of a fast reactor, in this case an SFR, sodium fast reactor, in green, and then a light water reactor, a thermal spectrum reactor, in red; both the sodium fast reactor and the lead-cooled fast reactor or metal-cooled fast reactor, so the neutron energy spectrum for an LFR would be similar to that shown in the green curve.

Note that fission neutrons are born in a range of energies centered around one million electron-volts or slightly higher, as shown by the blue arrow on the right of this chart. In the thermal reactor, these fast fission neutrons are moderated or slowed down to very low or thermal energies, where thermal fission takes place around or below an energy of about one-tenth of an electron-volt. Note that this is an energy reduction of about seven orders of magnitude.

So, in thermal reactors, such as light water reactors, most of the fissions occur around the one-tenth of an electron-volt thermal peak that's shown in the red bump on the curve at the left. In fast reactors, such as lead-cooled fast reactors or sodium fast reactors, neutron energy moderation is avoided and fissions occur mainly in the fast energy range in the peak, at the right side of this chart.

To delve into this a little further, consider the graph on the left of this viewgraph. This is a plot of fission cross-sections for U-235 and Pu-239, the two most important fissile isotopes in either thermal or fast reactors. The term fissile means an atom that can be easily fissioned after absorbing a thermal or a fast neutron. And a fission cross-section with units of barns is a measure of the likelihood of fission given an interaction with a neutron.

On this graph you can see the vertical band of energies in blue, at the right side of the chart on the left, that represents the range of energies of neutrons that are born by fission. If you look to the right of the graph, you will see that for U-235 and Pu-239, the fission cross-sections at fast energies of around a million electron volts, or MeV, are about 1 or a few barns, if you read the scale to the left.

You also see that for U-238, there is a similar cross-section for fission at energies higher than about 1 million electron-volts, but essentially no chance of fission induced by neutrons at low energy. Now if you look to the left of the graph, you will see that in the low thermal energies, the fission cross-sections for Pu-239 and U-235 rise to levels around 1,000 barns, so fission cross-sections are nearly three orders of magnitude higher in thermal than in fast spectra

This in part indicates the favorable characteristic leading to widespread current use of thermal reactors, and it also indicates why the level of fuel enrichment in fast reactors are generally considerably higher than those in thermal reactors.

Next, consider the graph on the right of this slide. This shows fission cross-sections in red and capture cross-sections in blue, Pu-239. The curves are also similar for U-235. The ratio between these two values is very important; hence, fission is a process providing not only energy but also neutrons needed to sustain the chain reaction, while capture is a process that removes neutrons.

So, note that there is a sharp decrease in capture cross-sections at high energy, especially in comparison with the fission cross-section. This is where one of the main advantages of fast reactors come in.

The point is that neutron energy spectra have important implications for both fuel utilization and minor actinide consumption. The histogram on the left of this slide shows the ratio of fission to absorption for a number of important isotopes in reactor fuel. The red bars show this ratio for a thermal reactor, in this case a pressurized water reactor, and the blue bars for a fast reactor, in this case, again, a sodium fast reactor, again, noting that for a lead-cooled fast reactor the results would be quite similar to those with a sodium fast reactor.

The chart shows that the fissile isotopes, U-235 and Pu-239, are likely to fission in either thermal or fast spectrum. The fertile isotopes, for example, U-238, which is fertile in the sense that it can be converted to a fissile material, are more likely to fission in the fast spectrum, and this was also shown on the previous viewgraph, where we noted that U-238 has an appreciable fission cross-section, but only for neutrons with high energy.

Note also that higher actinides, for example, plutonium, neptunium, americium, and curium, which are responsible for much of the long-term radiotoxicity of spent nuclear fuel or high-level waste, are much more efficiently consumed in the fast spectrum than in the thermal spectrum.

Now consider the chart on the right, which shows the number of neutrons per fission as a function of the neutron energy, and you can see that for fast energies, around 1 million electron-volts, the yield begins to climb considerably, so there are more excess neutrons available in the fast spectrum. The net result is better fuel utilization and significant actinide consumption in fast reactors.

So what does this mean for sustainability of fast reactors? This chart shows the results of a recent set of calculations by a colleague of mine, Dr. Luciano Chinotti, to consider the potential impacts of fast reactor scenarios in the United Kingdom. The question was: what would be the annual nuclear material requirement needed to produce 100 TWh of electricity or 30% of the total annual electricity demand in the UK? Three scenarios are shown, the first being the scenario using thermal reactors without recycle of spent fuel, labeled on the slide as the baseline scenario, the second scenario being the use of fast reactors, namely, the lead-cooled fast reactor, with recycle of plutonium, but not the additional minor actinides, and the third scenario being fast reactors with full actinide recycle.

In the first scenario, 2,100 tons of fresh natural uranium would be required annually. Of this material, 1,900 tons labeled here with the letter (a), would be set aside as depleted uranium, and 184 tons labeled as (b) would be leftover as still enriched uranium in spent fuel. There would also be 2.6 tons of generated plutonium labeled (c), .38 tons of minor actinides labeled (d), and 13 tons of fission fragments accumulated in the spent fuel.

Shifting to the second scenario, the new natural uranium feed would amount to only 10.8 tons per year in an equilibrium cycle, a factor of about 200 less than the first scenario, or none at all if all the accumulated amounts from the legacy use of thermal reactors is considered. This would be the utilization of previously-accumulated materials labeled (a), (b), and (c).

Note that in the second scenario the new minor actinide generation is somewhat less than in the first scenario and also that the fission product waste is considerably less. This is due mainly to the improved efficiency of power conversion in the postulated fast reactor system.

The situation for the third scenario is similar to the second, except that the minor actinide recycle eliminates most of the minor actinide materials, which is an important consideration in the issue of high-level waste disposal.

The takeaway here is that fast reactors support enhanced sustainability of nuclear power relative to thermal reactors through greatly improved uranium resource utilization and a significantly reduced amount in radiotoxicity of high-level nuclear waste. It should be noted that sustainability and improved material management are not sufficient alone to drive successful development of new fast reactors. Economic viability, safety excellence, and proliferation risk management are also essential.

Up to now, we've been considering fast reactors in general, so now let's consider the lead-cooled fast reactor in particular. Why LFR technology?

As with other fast reactors, LFRs offer significant advantages in sustainability and uranium utilization, in other words, better use of natural resources. They also offer the potential for a dramatic reduction in the quantity and toxicity of high-level waste if full recycle or a closed fuel cycle is used, as shown in the previous calculation.

Relative to other fast reactors, lead-cooled fast reactors have a unique combination of favorable features, which include a very high boiling point. It's 1,737°C for molten lead, a relatively benign chemistry, namely, there are no rapid chemical reactions with water or air, a low vapor pressure, which enables operation at near atmospheric pressure, and excellent neutronic properties for fast spectrum operation. So these and other favorable features are inherent in the properties of lead, the lead coolant, and can be exploited through proper plant design.

A lead-cooled fast reactor design starting with the design of a sodium fast reactor and simply replacing the sodium with lead would clearly not be an optimal design for an LFR. These favorable features must be taken into account and exploited in the design process.

However, there are also challenges to address, and the first is corrosion potential, and this is the one that gets the most attention so I'll put it aside for now and come back in a minute. Other challenges that need to be considered include the high melting or freezing point of lead, which is 327°C, and this requires proper engineering to avoid lead freezing by maintaining a temperature margin above freezing point throughout the primary system.

Another challenge relates to seismic or structural considerations due to the high density and weight of the coolant. The way that this issue tends to be addressed is first of all through compact size, which serves to mitigate the challenge of the high mass levels. And in the second case, in some designs, seismic isolation is integrated into the reactor system design.

A further issue is the fact that lead is an opaque, high-temperature coolant, and this has implications for in-service inspection as well as other operational requirements, the fact that there are similar in-service inspection issues and solutions with the sodium fast reactor, which is also an opaque high-temperature coolant, so some of the methods developed for the sodium fast reactor will also apply to the lead-cooled fast reactor.

An approach for several LFR concepts is to emphasize and implement accessibility and replaceability of components, which can allow for periodic inspection out of the lead coolant.

Finally, there are newer acoustic methods that are being discussed, studied, and appear to work well for lead-cooled systems.

Coming back to the issue of corrosion, it's well known that lead and lead alloys are corrosive to conventional steels at high temperatures. Corrosion prevention can be achieved by operating at temperatures low enough to avoid such corrosion phenomenon. For this condition, current materials could then be used. The temperature limit for this is generally taken to be about 480°C.

A second approach is the use of advanced corrosion-resistance materials for higher temperature operation. New materials, such as silicon or aluminumenhanced steels, for example, alumina-forming austenitic steels and silicon-enhanced steels is one approach.

Finally, surface coating with corrosion-protective materials for higher temperature operations, mainly from materials associated with fuel cladding and steam generator components, or functionally-graded composite materials, a technology under development at MIT, also show promise. Coating is of particular interest mainly for fuel cladding or for heat exchanger tubes. An R&D qualification program for the use of such coatings is necessary to demonstrate mechanical stability, adhesion to the substrate, and so on, under relevant operating conditions, which can include neutron irradiation.

The key factor in most designs involves maintaining a metal oxide film on metal structures by controlling the coolant oxygen within a range that is below the concentration for lead oxide formation and above the minimum concentration needed for sustaining a protective oxide coating of the component surfaces. In general, these challenges are technical in nature and can be overcome through proper design, engineering, and R&D.

Current reactor designs being considered feature different approaches or even combinations of approaches to corrosion control, and in many case, multiple approaches are used depending on which component and what temperature range the reactor is designed to operate in.

This slide provides a few comparative details for selected liquid metal coolants. The first is lead-bismuth. This is a lead-based coolant for some LFR designs. Then lead, and then sodium. Lead-bismuth, which is also referred to as LBE or lead-bismuth eutectic, has a melting point of 125°C, and a boiling point of 1670. Pure lead has both a higher melting point at 327°C and a boiling point of 1737°C. Both of these lead-based coolants are practically inert in terms of chemical reactivity with water and air, and this has important and favorable implications for the design, safety, and economic potential of LFRs. Sodium is also included on this chart for comparative purposes. Sodium has a lower melting point but also a substantially lower boiling point. The high chemical reactivity of sodium with water and air stands in contrast with the characteristics of lead coolants.

So lead and lead-bismuth eutectic coolants provide promising overall characteristics, while sodium technology is more highly developed having receive much greater R&D attention over the past 60 years.

Having mentioned LBE, let me say a few words about the lead versus LBE choice.

First, lead-cooled fast reactors can be cooled by either pure lead or by the alloy mixture of lead and bismuth, LBE, or lead-bismuth eutectic. LBE is the alloy of these two elements, lead and bismuth, with a composition that's approximately 55% bismuth and 45% lead.

The major advantage of LBE over lead is that it has a much lower melting or freezing point, 125°C for lead-bismuth versus 327°C for lead, and this reduces the engineering difficulties and allows lower temperature operation. And those are the major advantages.

On the other hand, LBE in the presence of neutrons produces polonium-210 through the reactions shown in red on this viewgraph. Bismuth-209, the primary bismuth isotope, reacts with a neutron to produce bismuth-210, which is a beta emitter with a five-day half-life, that ultimately decays into polonium-210.

Polonium-210 is an alpha emitter with a 138-day half-life, and it's a potent and radiotoxic alpha emitter, and that toxicity is one of the concerns. And the second concern is that it produces a significant heat load in the coolant itself.

Bismuth is considerably more expensive than lead and its limited availability may inhibit large-scale deployment of reactors cooled by lead-bismuth eutectic.

And so there is a series of advantages and disadvantages for each of these coolant types that must be considered by designers. Note that each of the Generation IV International Forum reference designs, that I'll talk about later, feature lead as the coolant, but several other reactor designs being actively pursued focus on lead-bismuth eutectic.

To summarize this part of the discussion, lead-cooled fast reactors have the potential to excel in safety for reasons outlined on this slide.

First is the very high boiling point of lead, as I said before 1737°C, or for lead-bismuth eutectic for that matter, and this allow reactor operation at or near atmospheric pressure and it virtually eliminates the risk of core voiding due to boiling of the coolant.

Second is the lack of rapid chemical reactions between lead and either water or air. There are no energetic releases or hydrogen production from chemical reactions. The use of water as an ultimate heat removal fluid is conceivable should other heat removal systems fail.

Third, the thermal capacity of lead combined with the large massive coolant means that there is significant thermal inertia in the event of hypothetical accident initiators, and there are long grace times for operator intervention in the event of an upset condition.

Next, lead is an effective shield against gamma radiation and it retains iodine and cesium, as well as other radionuclides at temperatures up to 600°C or higher. This results in a significantly reduced potential source term in case of fuel failure and contributes to enhanced defense in depth.

Finally, the low neutron moderation of lead allows greater fuel spacing without excessively penalizing neutronic performance. This results in reduced risk of flow blockage and reduced core pressure drop, while enabling a simple coolant flow path to allow operational and decay heat to be removed through natural circulation.

This chart summarizes the results of a presentation by the well-known Russian scientist Prof. Georgy Toshinsky in which he calculated the stored energy in three different reactor coolants: water, sodium, and lead, or LBE. With his assumed operational parameters he calculated the total potential energy in gigajoules per cubic meter for each of these coolants, taking into account the thermal energy stored in the coolant, the energy of pressurization, the potential chemical energy of interaction with zirconium, water, or air, and the potential secondary chemical energy resulting from the interaction of released hydrogen with air.

The bottom line is that water as a coolant presents 21.9 GJ/m3 of stored energy, sodium less than half of that figure at 10 GJ/m3 and lead a further reduction of about an order of magnitude down to 1.09 GJ/m3 of stored energy. The very low comparative amount of stored energy in lead-cooled fast reactor coolants is another indication of their enhanced safety potential based on the intrinsic properties of the coolant.

Now let's shift to the Generation IV International Forum, or GIF. GIF was formed in 2001 by a group of nine countries to develop future-generation advanced nuclear energy systems. Subsequently, several additional countries joined the organization, and membership now stands at 14 countries.

In 2002, the GIF published a Technology Roadmap document, which is shown here at the right side of this chart. Six Generation IV advanced nuclear energy systems, as shown on this table, were identified as having good promise, and three out of the six systems, the GFR, the LFR, and the SFR, are fast reactors. In addition, the molten salt reactor and the supercritical water reactor also have considered fast spectrum options.

The 2002 GIF roadmap included projected timelines for the various development phases for each reactor type, and that is shown on the left of this slide. Note that the three phases shown by different colors indicate the viability phase in orange addressing the question, are there any showstoppers, the performance phase in yellow focusing on testing, and the demonstration phase in gray indicating design, construction, and operation of a demonstration reactor.

The graph on the right shows the projections as of an update to the roadmap completed in 2013. It's interesting to note the key transition from the performance phase, yellow, to the demonstration phase, gray. In each case, the timeline for this transition was extended going from 2002 to 2013, by five to ten years, except for the case of the lead-cooled gas reactor, which shows a one-year extension from the original projection. This was mainly the result of the Russian efforts to build the BREST-OD-300 demonstration reactor, which I'll describe later, scheduled for operation in 2021.

So you can see that the lead-cooled fast reactor is now the Generation IV system with the earliest projected demonstration. The bottom line is that LFR technology readiness is higher than generally thought.

With respect to the lead-cooled fast reactor status within GIF, a Provisional System Steering Committee was first formed in 2005. Its members included the European Union, the United States, Japan, and Korea. The committee prepared an initial draft LFR System Research Plan, or SRP. Then, in 2010, a memorandum of understanding, or MOU, was created and signed between the European Union and Japan formalizing the LFR Steering Committee.

In 2011, the Russian Federation added its signature to the MOU, resulting in a revision and augmentation of the System Research Plan. In 2015, Korea became a full member by adding its signature to the MOU, and at present, the United States and China participate in active observer status.

Within the SRP, there are three reference systems adopted by the committee, and they include, the European lead fast reactor, or ELFR, "elfer," which is a large central-station 600 MWe reactor, second the BREST-OD-300, which is a demonstration reactor of intermediate size at 300 MWe, and the SSTAR 20 MWe system, a transportable, small modular system, or SMR. I'll provide a brief overview of each of these reference systems in turn.

The first of these reference systems is the European lead fast reactor, or ELFR. This is a 600 MWe pool type, central station, lead-cooled fast reactor, which is cooled by pure lead at a coolant temperature cycle of 400 to 480°C. It features removable steam generator pump assemblies and operates at a power conversion efficiency of 42%. The fuel is fixed oxide fuel, and it uses two-dimensional seismic isolation to provide earthquake protection.

Please also note that associated with ELFR is a smaller demonstration reactor known as ALFRED, which would operate at 125 MWe.

Here you see a sketch of the ALFRED demonstration reactor, which is a scaled down version of ELFR. ALFRED has a power rating of 125 MWe, and in common with ELFR, ALFRED using the coolant temperature cycle of 400 to 480°C, as well as many other components and features in common with ELFR.

Next, let's consider the LFR initiatives in Russia.

It was Russian or perhaps more correctly Soviet military applications that led eventually to the current LFR developments in Russia, and this provided a very rich knowledge and experience base leading to its current commercial reactor initiatives. This background started as early as 1951 with testing facilities and led to the deployment of a series of submarines and ground-based reactors cooled by LBE. In all, this experience base amounted to about 80 reactor years of operating experience with many lessons learned along the way.

Ongoing developments in Russia include the two systems on the right, the SVBR-100 and the BREST-OD-300 reactors. The SVBR-100 is perhaps the reactor type most directly following the Russian submarine experience as it is a small reactor cooled by lead-bismuth eutectic. The BREST-OD-300 is the second reference system in the GIF program, which we'll take a look at next.

So the BRES-OD-300 is a prototype for a commercial LFR. It operates at a power of 300 MWe and uses pure lead as the coolant. It operates at a 420 to 535°C temperature cycle. The fuel material is uranium-plutonium nitride, and it has a power conversion efficiency of 43.5%.

A couple of additional notable features are the unique concrete steel reactor vessel, which creates a hybrid pool loop type arrangement. And the plan to associate a pyrochemical fuel reprocessing facility with the reactor. I'll also note that the BREST-OD-300 reactor is a demonstration system as a prototype, but that also there is a much larger follow-on system, the BREST-1200, which has been envisioned.

One of the first concepts for a small modular reactor or SMR is the small secure transportable autonomous reactor or SSTAR, which was developed in the US by a team of national laboratories and universities.

On this slide are some sketches and article headlines describing early versions of the concept well before the current global interest in SMRs emerged.

The SSTAR concept is now a legacy design, work having concluded on it several years ago. It has been retained as a GIF reference system to represent potential SMR applications. SSTAR is a small natural circulation LFR of 20 MWe output operating on a 420 to 567°C temperature cycle. It uses nitride fuel in which the nitride is enriched in nitrogen-15. It features the use of natural convention coolant circulation for both operational and shut-down heat removal.

Power conversion is by a supercritical CO_2 Brayton cycle, and this provides an efficiency of about 44%. The concept envisions a long-life sealed core in a small transportable system.

So as a recap, this table summarizes some of the key parameters for these three reference systems with thermal and electric power levels, primary system types, coolant temperature cycles, power conversion and efficiency for each of the reference concepts. The intent of including these multiple concepts as reference LFR systems is to represent a full range of power ratings and application types. This table, as with the rest of the presentation, will be available for download following the webinar.

In addition to the three reference systems, there are several other concepts being currently considered or developed by commercial interests, laboratories, and universities internationally, and this is an indication of the diverse international interest in LFR technology and the potential for innovation and design.

This slide shows a selection of these initiative including: the Hydromine AS-200, a 200 MWe system being developed by the US company Hydromine in concert with their design team in Italy, the LeadCold SEALER reactor originated in Sweden and being developed in Canada, CLEAR-1, a 10 MW system being developed in China, URANUS, a Korean design out of Seoul National University, and the Westinghouse LFR. We'll now take a quick look at each of these concepts before concluding.

The Hydromine AS-200 is a highly compact 200 MWe LFR, where the compactness has been achieved mainly by creative design and elimination of components traditionally included in metal-cooled fast reactors. For example, the concept is four times more compact than the Superphénix sodium fast reactor, two to five times more compact than other more advanced, metal-cooled fast reactors that are in the design space now.

It uses oxide fuel, a 420 to 530°C temperature cycle, and the coolant is pure lead. The Hydromine AS-200 was first publicly presented to the international community at a symposium held at Imperial College in London, in July of 2016.

The LeadCold SEALER reactor is a very small 3 to 10 MWe lead-bismuth eutectic-cooled reactor being developed for applications in Canada and elsewhere, initially intended for commercial production of electricity in communities and mining operations in the Canadian Arctic.

It uses oxide fuel and the low-temperature operation, below 450°C, with a long core life of 10 to 30 years. It was designed to be the smallest possible reactor core that could achieve criticality at a fast spectrum using 19.9% enriched uranium oxide fuel.

It's a system that would be transportable to and from its operating site. Earlier this year, LeadCold received $200 million in funding from the Essel Group, Middle East, to enable licensing and construction of a demonstration reactor in Canada.

The CLEAR-I reactor under development by the Chinese Academy of Sciences is a very small system of 10 MWth power, cooled by lead-bismuth eutectic. It's designed to operate in a subcritical accelerator-driven mode or as a critical system.

It's fueled by uranium oxide and uses natural circulation for operational and shutdown heat removal. Its temperature cycle is the very low 260 to 390°C. Thus far, a detailed conceptual design of CLEAR-1 is complete and the preliminary engineering design is under way.

The Korean URANUS reactor is a 50 MWE lead-cooled system using natural circulation cooling and operating on a temperature cycle of 400 to 520°C. URANUS features 3D seismic isolation, underground siting, and a 20-year refueling cycle. And as I said before, it's being designed and developed at Seoul National University in Korea.

The Westinghouse LFR is a concept under active development and for which I can't provide detailed specifications beyond the sketch here. However, it's worthwhile to point out that this initiative is the result of a comprehensive independent evaluation several years ago of next generation advanced reactor technologies.

Westinghouse selected the LFR as the technology having the greatest potential to meet key requirements of safety, economics, and marketability. Also considered was sustainability and technology readiness, and a clean sheet approach was used. There's no legacy from the past.

Some of the key elements of their assessment included the potential for plan simplification from atmospheric pressure operation, the fact there are no significant sources for containment pressurization, and there are no boiling concerns, a strong safety case, and sufficient technology readiness.

Their assessment identified several favorable economic indicators resulting from the enhanced safety of the LFR systems, including, first, the expectation of reduced capital cost from the plant simplification based on a reduced number of components from a primary system operating in atmospheric pressure, the potential elimination of an intermediate circuit, the small and easier or faster to build containment due to the lack of significant sources of pressurization, and the lack of a need for special provisions, systems, and components to protect the plant from coolant leakages and coolant-water or air interactions.

Second, the expectation of high plant efficiency. The large margin to boiling makes LFR efficiency dependent on progress and materials and therefore higher temperature operations rather than on coolant boiling concerns.

Third, the high power density from the use of a liquid metal coolant.

And finally, a strong case for a reduced emergency planning zone based on a reduced resource term as a result of the large margin to boiling, a high thermal capacity, reduced likelihood for a loss of coolant for LOCA, loss of coolant accident, and the use of a chemically benign coolant coupled with lead's ability to retain radionuclides.

So in conclusion, there's a clear growing international interest in LFR technology. Some factors for this include: excellent sustainability from full utilization of uranium resources; reduced nuclear waste concerns due to the ability to consume minor actinides and utilize accumulated plutonium as fuel; an outstanding safety case; and promising economics from lead's inherent attributes combined with proper design. So these are some of the main drivers of this international interest.

I leave you with this table for future reference, which summarizes the array of potential operating and design parameters of LFR systems, describing some power-related characteristics, thermohydraulic parameters, and materials. And this should remain available as part of the downloadable presentation material.

And similarly, this is a shortlist of some selected reference materials, which will also remain available in the downloaded webinar material.

And with that, I will conclude my presentation. Thank you, all.

Oates: Thank you, Dr. Smith. We appreciate your time in pulling together these slides. Again, we apologize for the technical difficulties with our audio and appreciate you hanging in with us to make your presentation.

Amanda, if you are on, I have been dropped from the host. I need to be elevated. I cannot see the questions coming to help with the feedback. If you have questions for Dr. Smith, you can post them in the Q&A pod and we will take as many of those questions as we have time right now.

Upcoming webinars, in July, Dr. Michel-Sendis with OECD-NEA from France will present on the Thorium Fuel Cycle, in August, a presentation by Dr. Steven Hayes with INL in the USA on Nuclear Fuel and Materials, and in September, Energy Conversion by Dr. Richard Stainsby.

So I don't see questions. I do see a comment. Craig, if you toggle between the Q&A pod between the presenter and the participant view, you'll be able to see those questions come in. The first question, what is the performance difference between UO2 and UN fuel in the LFR?

Smith: Okay, yes, uranium oxide is a fuel that is commonly used for thermal reactors, for light water reactors, but also for fast reactors, for sodium fast reactors. It's considered a fuel that's very, very well known and very well characterized.

Uranium nitride fuel is a fuel that has some characteristics that are beneficial both in terms of the operation of the reactor, namely, it has superior thermal connectivity characteristics that help in the thermal-hydraulic performance of a reactor.

And then secondly, for purposes of reprocessing or recycle of spent nuclear fuel, there are some significant advantages with uranium nitride fuel in moving from kind of the wet chemistry fuel reprocessing to pyrochemical processing.

So both oxide fuels and nitride fuels are incorporated into the designs of LFRs. I would say that the nitride fuel is considered to be more of a fuel for higher temperature operation in the future, and oxide fuels are the safer bet in terms of fuel qualification to remove obstacles for near-term development and deployment.

Having said that, I would point out that the first demonstration reactor in the Gen IV program is the BREST-OD-300 reactor, which is fueled by a nitride fuel, so nitride fuel is not necessarily that far out in the future. It's one that will be appearing in the early deployment of the LFR.

So the second question here has to do with silicon carbide materials which are receiving attention for the advanced light water reactor and are being researched extensively. I do think that there is potential for lead or LBE systems of silicon carbide, and I understand that this is a material that is being considered for some applications. And again, it's probably a material which is further out in the future, but getting to materials that can operate in a higher temperature regime is very interesting and important because, as I said in the presentation, the obstacle to higher temperature operation lies in materials for the lead-cooled fast reactor, unlike other reactors where margin to boiling might be a bigger consideration or increased pressurization. For the LFR, I think as you can move to higher temperature operations, you will improve performance, and materials is the key.

Another question is, will the higher temperature improve electricity production?

I think the real answer there is that there's a direct relationship between the temperature of operation and the efficiency of power conversion, and so by improving the efficiency of output you improve the economics and the quantity of electrical output for a given amount of fuel that you consume. And so I think the answer is that, yes, this would be a significant improvement and it really indicates an upside potential.

In the one calculation that I showed comparing thermal reactor systems to fast reactor systems for the UK, you noted that the quantity of fission product residue was dramatically reduced in going to the LFR, and that's primarily due to the higher temperature operation and the improved deficiency going from some efficiency in the low 30s-percent up to the low 40s, and that can have a significant impact in the efficiency of the operating system.

So, the purity level for nitrogen-15, I think there are two nitride fuel systems that I've talked about, and the use of nitrogen-15 is included in the SSTAR reactor for purposes of improving neutron performance, the neutronics, so the level of purity does not need to be extreme. And I'll say that on the one hand, but then I'll also say that the Russian BREST system with nitride fuel uses a normal isotopic mix of nitrogen. And so it's basically a design choice that is made in looking at nitride fuels, whether you go with the isotopically-enriched version or the conventional version. You can do either one and there are pros and cons with each.

So the Russian... The question is, are the Russians still using ferritic steel, stainless, with oxide coating, not ASA steel?

So I don't have the up-to-the-minute answer to that question, but the Russian design relies heavily on the control of oxygen content in the coolant to maintain the oxide coating on their components, and I believe it's a silicon-enhanced steel that they're using, but they rely very heavily on oxygen control.

And then I would say that there are other systems that are being looked at where oxygen control is not the primary approach. If the temperature is kept low enough, then one can use a coolant that has very little oxygen in it because the corrosion processes don't really kick in until you get above a certain threshold.

I think that's the last question I have. If others have questions after the seminar, after the webinar, please feel free to shoot me an email. I'd be happy to open a discussion.

Oates: It looks like there are two more questions.

Smith: Yes. So there's a question that says, have sufficient tests been made on nitride fuels?

I know that nitride fuels, the fuel concepts for the BREST, the Russian BREST system, are ongoing, and they are in the process of actually testing their nitride fuels. And they're on a very tight schedule, as I indicated before. Their current stage is that they have completed their design and they are awaiting construction approval from their regulatory authorities as we speak, with the plan to begin construction and then operation by 2021. So they will have completed all the tests that they deem necessary on nitride fuels and those tests are ongoing.

Okay, so the question here is, the distinction between built technologies and plans from proposed evolution of designs, and asking for the referenced LFR concepts, are there concrete plans and funding?

So there are three reference designs that I had mentioned. One of them is the BREST system, which I just described the current status of that project. It's certainly very concrete with funding in place and at the verge of going forward with completion of construction based on their full design and so on.

In the case of the European lead fast reactor, it's more preliminary and not all funding is in place. It has been funded over the years incrementally, and there is an active effort to move forward with this project, particularly the demonstrator, which would be, as my understanding is at this point, which would be sited in Romania, and I understand that there is either funding lined up or the expectation of infrastructure funding to support at least part of that.

So I would say that with the European lead fast reactor, it's in an intermediate state where not all funding has been secured, but the design work has continued and it's seeking funding to firm things up.

And then with the SSTAR small reactor I mentioned, that's kind of the other extreme, and as I said in the presentation, it's a legacy system. It's a system that went through conceptual design and then basically was put on the shelf, it's adopted as a reference... [*sound cuts out*]

Operator: Please hold while I confirm your passcode. Thank you. Your passcode is confirmed. When you hear the tone, you will be the fifth person to join the meeting.

Oates: Craig, we dropped off that audio mid-answer. You got up to about the SSTAR reactor, and I apologize when that notification popped in.

Craig: Okay, do you want me to go back and reconstruct that?

Oates: Let me make sure that we are broadcasting. And we are. So, yes, if you want to, it was the third reactor I think when we kind of got cut off.

Craig: Sure. So I understand that the sound was cut off on the response to that last question partway through. I had mentioned the status of the BREST-300 and ELFA reactor, and I began to say that for SSTAR, which is the third of the three reference systems, the status is, as I mentioned in the presentation, that this is a legacy system. It's a system that the initial work of conceptualizing the design was completed and then the design put on the shelf, and further development has not been continued on that system, and it occupies a place on the GIF list of reference reactors as, I guess, more of a placeholder than anything else to basically represent the space for small modular reactors, which is recognized as a promising segment for GIF systems.

So that's the three reference systems. They're in very different stages of development and commitment of funding. I would say that in the additional systems that I mentioned, each one of the them has differing levels of design teams that are very active in pursuing the respective designs and with the appropriate funding to carry out their design activities. In particular, LeadCold reactor received an infusion of funding intended to support its efforts to carry it through finalization of design and demonstration in Canada, a $200 million investment in that concept, and so that one maybe stands out in terms of securing funding to carry that additional system forward.

But I would say the systems in Korea and in China, the Hydromine system and the Westinghouse activity are very active with design teams working at it and with the appropriate level of funding secured to continue their design activities. And I hope that answers the question.

Oates: Great, thank you. I don't see additional questions. If there are any other questions, go ahead and type them in now, and again, my apologies for the technical issues this morning. I do appreciate everyone's patience as we work through those setting up another meeting room. These things happen periodically. Thank goodness it's not very often. We've had quite a run of successful webinars without technical issues.

This presentation was recorded and will be posted on the GIF website with the slide deck as PDFs for future reference, and if there are no additional questions at this time for Dr. Smith, then I think we'll conclude the presentation and wish you all a good and safe day. And thanks again, Dr. Smith, for your marvelous information.

Smith: Thank you very much.

Paviet: Thank you, Craig. Bye.

Smith: Thank you, Patricia. Bye.