Sodium Cooled Fast Reactors (SFR)
Dr. Bob Hill, ANL, USA

Berta Oates: Good morning, everyone. Welcome to the next Gen IV International Forum webinar. We're just getting started. Today's presentation is on sodium cooled fast reactors. I'd like to take this opportunity to thank all of you for attending and to introduce Dr. Patricia Paviet. Patricia is the Materials and Chemical Technologies Director within the Department of Energy Nuclear Energy Office, and she is also the Chairperson for the GIF Education Task Force responsible for bringing you these webinars. Without any further delay, I'll turn the time over to Patricia.

Patricia Paviet: Thank you so much, Berta, for the introduction. I am very, very happy to be here today, and I'm very excited to listen to Dr. Bob Hill's presentation. Bob is a senior Nuclear Engineering at Argonne National Laboratory, where he has worked for the last 29 years with research focused on reactor physics, fast reactor design, and fuel cycle applications.

Dr. Hill completed his PhD in Nuclear Engineering at Purdue University in 1987. His current position at Argonne is Technical Director for Advanced Nuclear Energy R&D. He has previously led Nuclear Engineering Division research groups working on reactor physics analysis, advanced modeling and simulation, fuel cycle, and system dynamic modeling, criticality safety, and nuclear data.

Bob is co-National Technical Director for multi-Laboratory R&D activities in the DOE Advanced Reactor Technologies Program; this work includes small modular reactors, advanced structural materials, energy conversion technology, methods validation, non-LWR licensing, and system integration. He also serves as the US representative for the Gen IV Sodium Cooled Fast Reactor collaboration. And I would like also to add that he recently became an American Nuclear Society Fellow.

So this is my great pleasure to have Bob presenting this webinar, and, Bob, thank you again for your dedication, and I give you the floor. Thank you, Bob.

Robert Hill: Patricia, thank you for the introduction, and I'd like to welcome all the participants to the webinar.

I'm going to cover three topics in the talk this morning that I'm going to address related to the Generation IV, some work on sodium cooled fast reactors. First, I'm going to spend a little bit of time on the basic physics. This is important to explain the missions and motivations for fast reactor development. Secondly, I'm going to give a technology overview. This will include a history of SFR experience internationally. And lastly then, I'm going to give a description of the collaboration that we have in the Generation IV International Forum on SFR technology.

This first viewgraph is a short review of the some of the material that was presented at the last webinar on nuclear reactor design. This is a picture of the energy of neutrons within the reactor system. The fission reactions which produce the energy in a nuclear reactor create neutrons at high energy. This is shown in the figure here. Neutrons are born with a high energy. And then as neutrons interact with the materials in the reactor system, they scatter with the materials in the lab reactor system and they tend to lose their energy, and so they move down in energy as they are moderated, which means they lose their energy and slow down by neutron scattering reactions.

Most of the interactions and the creation of fission reactions and the capture of these neutrons in a thermal reactor system happens down here when the neutrons have lost nearly all of their energy, and that's the behavior you have in a thermal system with the reactions happening down here.

Conversely, in a fast reactor system, you avoid the materials that can moderate the neutrons or slow them down, and you have most of your nuclear interactions happening in this energy range, fairly close to the energy at which the neutrons were created.

This behavior is important because the nuclear interactions and the types of reactions you have are different in this energy range than they are in this energy range. I'm going to show that for one particular isotope here. This is for Pu-239, which is one of the dominant fission isotopes, especially in fast reactor systems, and this is the energy range for fast reactors here, near where the neutrons were created, and then this is the range in which you have most of the interactions in thermal reactor systems.

Two major differences that you can see off of this. First of all, the probabilities of reaction are much higher, about three orders of magnitude higher here in the thermal range where the neutrons have slowed down and lost their energy compared to the fast range. The other important behavior is that the red in this curve is the fission cross-section and the blue is the capture cross-section. You can see in the fast energy range here, where the neutrons have not lost their energy, that the fission cross-section is much higher than the capture cross-section.

These differences in the fission-to-capture ratio, as well as the overall difference in the probability of fission happening, lead to very important differences between what we call the fast reactor systems, which have neutrons in this energy range, and the thermal reactor systems, which are dominated by neutrons in this energy range.

This behavior with the fission being dominant in the fast spectrum, you can see this for the Pu-239 in this particular chart, which is the fact that you have a much higher fission fraction probability in a fast spectrum, it's up around 85%, compared to around 65% in a thermal spectrum for the Pu-239.

You see similar behavior for the other actinides. You can see, there's a comparison here from U-235 up into the curiums actually, where you have higher fission fractions in the fast spectrum than you have in the thermal spectrum, which is the blue versus the purple color here. This difference is especially pronounced in the even isotopes, which to have no fission in a thermal spectrum system like Pu-240, compared to the fast spectrum where you actually can get 50% fission in the Pu-240.

This behavior and this difference in physics between the two energy ranges has important implications for the fuel cycle behavior. Those come from two particular aspects of this dominance of fission in the fast energy range.

The first is that you have a better neutron balance in the fast energy range. A better neutron balance means you're more likely to fission in the fissile isotopes than you are to capture which does not sustain this chain reaction. So you have a better neutron balance in the fast spectrum.

You also do not get as many capture reactions in the fast spectrum. You get fissions, not captures. The capture reactions actually can take, and from the initial uranium starting material and they capture you up into the plutonium, but then also up into the higher actinides. And the generation of these higher actinides, which would be the americium and the curium cause difficulties….

I think I just lost my connection. I'll be back in a moment…

Sorry about that delay. I briefly lost my connection there.

So you have the two aspects here from the difference in the fission fraction and the fast range. You have the neutron balance and you have the fact that in a thermal spectrum system, this capture leads to these higher actinides. These higher actinides tend to be more radioactive than lower actinides; therefore, these are problematic when you're trying to do fuel fabrication. They lead to doses in the fuel fabrication process, which complicates recycle of the materials.

So the implications then of this difference in the physics behavior between the fast energy range and the thermal energy range are summarized here. Thermal reactors, because of the higher probability of fission in the low energy range, can be configured to operate on low enriched uranium, but typically you configure thermal systems to operate on once-through fuel cycles. This is because the neutron balance is not as good as the fast reactors so you would need a fissile feed. You'd need more fissile material being introduced in the system to sustain their reaction. And also, you would have this capture to the higher actinides I was talking about, which would be difficult to manage when you recycled the materials, and the more times you recycle in a thermal system, the more of those higher actinides that you would have.

Conversely, fast reactors have been designed and intended for closed fuel cycle applications because in the fast system you suppress the higher actinide generation and the favorable neutron balance gives you much flexibility in how you configure the system to be either a breeder or a burner of overall of the fissile material.

This behavior in the fast spectrum with the higher energy neutrons is not a new phenomenon. This was recognized very early on in the nuclear era. Fermi, actually very shortly after the first chain reaction, had talked about the need to have fast reactors, and especially if you wanted to realize the full energy content of the uranium, this was a very high priority early on. You can see the timeline here, and there was a lot of work from the '50s through the '70s on the development of breeder reactor systems in that timeframe.

This was driven by two aspects. The first one was there was an anticipated scarcity of uranium. Secondly, there was an expected quickly accelerating and expanding demand for nuclear systems, which would exacerbate the scarcity of uranium. So it was thought early in the nuclear era that this was going to be a constraint on the ability to use this energy source, so there was a lot of research and a lot of development work on fast reactors for this particular aspect, its ability to convert the fertile material into fissile material because of the neutron balance behavior.

There was not the growth in nuclear in the 1980s that was originally anticipated, and there were additional uranium resources that were found, so some of the urgency for this particular mission was reduced. We had countries like the United States then that went back to a once-through fuel cycle on that timeframe, whereas other countries continued to develop the closed fuel cycle technology.

There has been renewed interest in the last two decades in the fast reactor technology based largely on the recyclability, and the recyclability can be an important aspect for the waste management mission, as I will talk about later in this presentation. And fast reactors are also important in order to enable that mission to basically exclude the actinides, and particularly some of the transuranics, which are the actinides above uranium, from the waste streams.

This early expectation and the interest in fast reactors has been confirmed in the last decade by a variety of international studies. I'm showing here just some of the summary results, as well as the figure of the study process from a recent study that was conducted in the United States. This particular study was completed in 2014, and it was focused on looking at a comprehensive set of fuel cycle options. There was 4,400 different ways to configure the fuel cycle that were considered in this, and 40 different evaluation groups in order to look at the behavior of those different types of fuel cycle scenarios and configurations, a lot of work in this to develop a broad set of criteria and metrics, and also to look at a broad spectrum of the fuel cycle all the way from the mining of the uranium to the eventual deployment and steady state for long periods of time of nuclear energy systems.

For this talk, the key aspect from this study is the conclusions, which are listed down here in blue.

The conclusions were, if I want to get significantly better behavior out of the fuel cycle, then I need these types of features in my new fuel cycle system. And you can see all of these features down here are the features of a fast energy, fast neutron spectrum system. This spells it out at fast neutron spectrum but you want the ability to continuously recycle, so you need to have that higher actinide management behavior. You want high internal conversion for the resource extension aspect, and that again, you need the favorable neutron balance from the high energy neutrons, and the avoidance of uranium enrichment. Again, the neutron balances means I need uranium enrichment in thermal reactor systems.

So these recent results confirm that fast spectrum systems have a vital role if we're trying to improve the performance of advanced fuel cycles.

This just gives a picture of the fast reactor within the fuel cycle. Within this picture, the fast reactor would be located here, and so this would be your closed fuel cycle here where you have the material being irradiated in the reactor, then you have the separations to recover the materials for recycle, then you recycle, you fabricate the recycled materials, and then you go back into the reactor. So this is the closed fuel cycle here.

There's a lot of flexibility for fast reactors on how you want to operate within this closed fuel cycle. In particular, in this figure I'm showing the red dotted line here is the conventional approach to this, which is, you're using this closed fuel cycle to extend the uranium resources, so you go take the uranium from the ore, you by-pass the enrichment step, it becomes the makeup feed in the recycle fuel fabrication, and you're able to operate in this closed fuel cycle and extend the uranium resources.

You also can configure the fast reactor to be a net consumer and not a sustainer of the species and the actinides within this closed fuel cycle, and in that case, you could run the LWRs which are creating some of that material. You could again recycle the material and you could operate then within this closed fuel cycle with the makeup feed being the recovered materials from the other types of reactors.

This has been a fuel cycle that has been looked at in particular in the United States, but elsewhere also, and the purpose of this approach is primarily to exclude the actinides from the waste stream. By doing this instead of direct disposing, you're able to get a very different set of waste characteristics because the actinides are now going back into the fuel cycle and being excluded from the repository.

So you have two major approaches that you're taking here, major strategies for the actinide management. One is for the resource extension and the other is for the waste management aspects.

There are other favorable features of fast reactors. You'll see from some of the performance characteristics, they tend to be very high-power density systems, which can be very favorable for small reactor applications. Because they've been configured to operate in this closed fuel cycle, they've been designed to operate on plutonium and other fissile materials than just enriched uranium, so they are very favorable for plutonium management which is being looked at in several countries around the world.

Another aspect then, and what we're focused on in Generation-IV, is working on the reactor technology within this system to get the economics and the costs down to where they are favorable that you can use this as an electricity production system.

I wanted to go a little bit into some of the waste management benefits and discuss some of the aspects of it and how much improvement we're talking about in some of the fuel cycle behavior. So I'm going to look briefly at the uranium utilization aspects, and this is one of the benefits of the fast reactor technology is the ability to vastly extend the uranium utilization.

The upper part of this figure is showing the uranium utilization for once-through systems. In once-through systems, I first have to enrich the uranium, and much of the uranium then ends up being depleted uranium in the enrichment process, and then I only get partial consumption of uranium when it's in the reactor.

So for instance, a conventional PWR is going to be about 5% burnup and it's got about 4% enrichment. The net uranium utilization from that once-through fuel cycle then is just over half a percent with most of the uranium being in the enrichment tails, and then some of it being in the discharged fuel from the reactor.

You can see here, for thermal reactor systems, you can get to significantly higher burnups for some of the gas reactor, or maybe even extending the LWR burnups, but to do that, you have to go to higher enrichments. This is because of the neutron balance issues in a thermal spectrum. In order to get the higher burnup, I need to have a higher enrichment.

The net aspect on the uranium utilization, when I go to a higher enrichment, I have more material discharged and lost in the enrichment process, and the overall, the net utilization of the uranium, is very similar to what it is for the current LWR operating conditions.

So to get significant extension, what has been looked at is the recycle of the materials. You can get a little bit of an increase by recycling the LWR materials, but if you want to really use the uranium you need to go into a fast spectrum system where you're running it in a converter mode and you're basically taking that depleted uranium and you're using utilizing [*sound interference*] to fissile material.

Was there a question on being able to hear me? Okay, I'll continue then.

With that and with the recycle, you're able to extend the uranium utilization from the 1% range that we have in once-through systems to greater than 90% with the limitation only being what do I lose in the fuel cycle process in the separations and fabrication steps.

So this is a very large difference between 1% uranium utilization once-through and over 90% with recycle. There's been a lot of recent work, and this was asked about in some of the previous GIF webinars, on what you could do in between this range. With fast spectrum systems, you can get between the range of 1% and 90% with either an optimized once-through type system or with a limited recycle system.

I'm going to spend a little bit of time talking about some of those new options that have been looked at for some improvement of uranium utilization but not the full improvement that you get by going to a repeated recycle system.

The systems that are trying to extend the uranium utilization in a once-through or limited recycle mode are relying on breed and burn principles, and this viewgraph explains what those principles are.

So you would have an initial region, which has fissile material, either of uranium or plutonium recovered from elsewhere, and you're surround that by a blanket. Now a blanket in these systems is depleted uranium. So really this is depleted uranium with the intent for the blanket to capture the neutrons and to convert them, as talked about here, to convert that depleted uranium into fissionable fuel, with the fissionable isotope there being Pu-239.

This behavior and this internal conversion – it's called internal because it's within the reactor – is used in standard breeders. In standard breeders you have the blankets and then you recycle the blankets, recover their material and put them back into the reactor. The difference in these breed and burn concepts is you're trying to do that same physics and you're capturing the neutrons and putting them and creating the fissile material, but you're trying to extend the amount of that destruction and then fission of those fissile isotopes in the reactor. That would minimize the reprocessing and it would extend the uranium utilization without having to do repeated recycle.

Now there's been a wide variety of concepts and I'd be happy to answer questions at the end about some of those concepts to do this and to do the breed and burn, but the simplest one to envision what's being done in these breed and burn concepts is the CANDLE concept. This was a concept that was developed in Japan. And you can see here, it's a very simple configuration here. This is the starter zone, it's down here in the lower region. So basically, this is the reactor at the beginning of life. This is where all the neutrons are produced.

And then the neutrons that are produced in this region are leaving this region and leaking, and then they are leaking into this depleted uranium zone that is placed above the initial starter zone, and as enough neutrons leak into this depleted uranium region generate enough fissile material, being Pu-239 in this depleted region, that it becomes the region that's sustaining the chain reaction, and the power moves up from the starter region into this depleted uranium region where I've generated the fissile material. This is why it's called the CANDLE concept because it slowly moves upwards and then depletes and the fissile material goes out.

So this shows the behavior of the power density. At the start of life you have this peak down at the very bottom. This is core height. And then the peak of the power moves with time. This particular example was actually for an eight-meter-tall system and it went for 200 years where we started with the starter without putting anything else in besides the initial depleted uranium. It slowly crept up, and then it eventually depleted all the material with the power ending up at the top end of the system.

So to compare then the performance of that type of system to some of the more conventional recycle systems and to LWRs in this table, you have on the left column here LWR systems. You have here both conventional burnup, which is around 50 GWd/t and a high burnup option here, around 100. On the right side here I've got a conventional fast reactor system where I'm doing recycle. And then I have in the middle the behavior of one of these breed and burn concepts, in particular the CANDLE concept here.

So there are two large differences in the performance that are highlighted there. The first one there is, because you have this very large depleted uranium zone that you have put into the reactor at the start of its life, you have a much higher heavy metal inventory for these breed and burn systems than you do for either LWRs or for conventional recycle systems.

But you are over time able to slowly consume that. You can see here a cycle length for these systems where every two to three years I have to put new material into the recycle system, or the LWR. For 200 years this system just sits and slowly consumes the material, and you're able to get significantly more of the material consumed. You're able to go to a burnup here, which is about two-and-a-half times that for even the high burnup version of the LWR.

Now in the conventional recycle system, you only went to about 10% burnup each time through, and then you took the material out and you put it back in to take it to a higher burnup the next pass.

So the difference in this behavior then on fast spectrum systems, you're able to get a once-through about 25% destruction, so you aren't able to get up to the 90% you can get with the recycle, but you are able to get a significant extension beyond LWRs, up into the 25% range. If you had an even higher fuel burnup limit, you could probably get up to 40 or 50%, which is talked about in some of the concepts that are being used and looked at internationally.

However, when you're operating in a fast spectrum system and you're trying to go from this burnup here, which is about 10% up to 25%, that implies longer resonance times and a lot more neutron and radiation damage than you have in conventional fast reactor technology.

So the challenge for these breed and burn type techniques to get some extension of the resources without having to do recycle is they will require materials that can go to higher neutron irradiation and can sustain the damage because the reason that you have only this 10% burnup in a conventional recycle system is because you've had enough fast neutron damage that you need to take out and reconstitute the materials at this point. But if you can develop those types of materials, you can go to this type of behavior with a once-through or you could extend the burnup in the conventional recycle, but you would still in a conventional recycle probably do the recycling to get the full uranium utilization.

Next, I'm going to turn to historical perspective on the fast reactor technology development, and even a step back from that, to start with, there's a lot of different coolants, an extremely wide variety of reactor coolants, that were tried in the early development of nuclear technology. I have the full list here, and almost all of these examples here, there was at least one country that did at least one critical assembly looking at this type of coolant, although some of this was out of pile testing also.

But water coolant, we know this has been commercialized with the light water commercialized in many parts of the world and the heavy water commercialized in Canada and also elsewhere.

There's a wide variety of liquid metals looked at. You can see the long list here. Some of these were not continued, like mercury, because you were worried about some of the toxic material issues. Others had way too much neutron capture to be useful in practical systems.

You had gas technology, which was developed and used extensively in the United Kingdom in particular.

You had a variety of fluid fuel concepts. We've heard a lot of talk about molten salt lately, but you see an even longer list here of some of the things that were looked at on the fluid fuels.

And you had organic coolant technologies. There was a reactor that was operated briefly with kerosene as the coolant for that reactor system.

You also had a look at different combinations of coolants and moderators. Some examples of that are the Hallam reactor, which was graphite-moderated but sodium cooled. So you took some of the technologies that were being looked at as coolants themselves, and you actually had hybrid systems here with different combinations of them.

For fast reactors, fairly early on the focus became on sodium and on liquid metals, and particularly on sodium. EBR-I actually used potassium chloride, and by EBR-II you were to the sodium technology, and as you'll see, most of the experience has been with sodium technology.

Some of the features of sodium that led to that are listed here. This shows, the check/pluses here are the very favorable qualities of the sodium coolants compared to other options in particular. With sodium for a fast spectrum system, it is a heavy enough element that it does not significantly moderate the neutrons, so you're able to operate in a fast spectrum system. It does not have a lot of capture so you don't lose a lot of neutrons to captures in the coolant itself.

The other aspects of this were it has very good thermal and physical properties and an ability to remove heat. This allowed you to go to high power densities. Compact systems, which you want to have, and I'll talk about briefly, for a fast spectrum, and to do that with low pumping powers because this is not a heavy coolant where I'm going to have to have a lot of pumping in order to push it through the reactor and to remove the heat.

In addition, the boiling point of °sodium was high enough. It's around 900°C that you could operate on conventional steels and you could have a significant margin to melting. As you'll see, most of the operating reactors on sodium coolant have temperatures around the 500 to 550°C range which is where you want to be and where you were limited by the structural materials of that time with the metals, and that gives you a very significant margin to boiling still.

This is a list, and I apologize for the busy-ness but part of the point here is to communicate that there have been a lot of different test and demonstration reactors internationally. This is a list of all of the different test and demonstration fast reactors that have been operated. You can see from the last column here, as far as coolants, that nearly all of these have been sodium. You had some early experience with potassium chloride and with mercury, but since then, and the only experience that wasn't with sodium was the Russian submarine reactors, which used lead-bismuth as coolant.

You can see in the development of these tests and demos, in general you have a scaling up of the size. These were some very small systems, starting off with the 1 MWth range up to the largest being in Superphénix and the 3,000 MWth range. Several of these reactors are operating today. Those are the ones that are in red or ready for operation today, and you have several new projects that have either been started or reactors that are almost complete, like the Indian prototype reactor.

Some of the key experience over that time and some of the key accomplishments from that are the first electricity that was generated by nuclear reactors was from one of these fast test reactors, in particular EBR-I; that was back in 1951. We then had operations and we had fuel cycle demonstration in the 1960s, which would be the EBR-II reactor. I showed the list on the previous page on the 20 fast reactors, and roughly those have about 400 operating years of experience, again, almost totally with sodium technology, besides some of the early work and the Russian submarine work.

You do have two power reactors operating today. BN-600 in Russia has operated since 1980 with about an 80% capacity factor over its 35-plus-year life. BN-800 was started up last year. It's about 880 MWe. And the Indian reactor, 500 MWe, should be starting up shortly.

The net result of this experience is the basic viability of sodium cooled fast reactor technology is demonstrated. The ability to build these reactors and to operate them with good reliability has been shown. The picture up here in the upper left, this is actually within the BN-800 vessel. This was about a year before they actually started up the reactor, just to show that there is modern construction and there is both the historical experience and recent experience with sodium technology and sodium cooled fast reactors.

Next, I'm going to give a bit of an overview on some of the particular facets of the sodium technology.

One of the key things that you'll hear talked about is the configuration of the sodium system. There are two different types that are being looked at internationally. There's the pool configuration as shown here, and there's a loop configuration.

In the pool configuration, this is similar to some of what they would called the integrated LWRs that are being looked at. The key aspect here is that the primary pumps and the intermediate heat exchanges are all located here, and I'll talk about the intermediate heat exchangers in a moment, they are located inside the vessel, and so this is all submerged within a pool of sodium. So the pumps then actually take the sodium and they push it back through the reactor and through the core here, and then it comes out and it goes into the heat exchangers once it's hot up here, and then goes back down into the pool, and then the pumps take it from the pool and put it back into the reactor.

This type of a configuration is what was used in Phénix and Superphénix. It also was used in the Russian BN-600 and 800 reactors. Some of the advantages of this approach are you have a vessel here with a significant heat sink with the sodium in it that has very few penetrations because you only have to take out the secondary sodium that's coming out in order to go and use it to create the heat for the system.

The other aspect here is that all of the primary sodium stays within the vessel, so you don’t really have to worry about leaks on the primary sodium because if you had a minor leak in one of these components, it would just leak the material back into the pool.

The other configuration that's being looked at in international designs is a loop configuration. This is a picture of the Japanese JSFR design, which is a loop configuration. In the loop, the pumps and the intermediate heat exchangers do not sit within the vessel; they are outside the vessel. And you have pipes then taking the primary coolant from after it comes out of the core and going through the intermediate heat exchanges, and then you have the pumped material then coming back into the system and then going up through the reactor core.

This configuration is similar to what is utilized in the Monju reactor and in the FFTF that were previously built.

Some of the benefits of this approach are you would have, because the pumps and the heat exchangers are not inside the vessel, the vessel will have a smaller diameter and be significantly smaller than it is for the pool system. You also have more ready access to the pumps and the intermediate heat exchangers because they're not submerged within the pool of sodium any longer, so you have a bit more freedom in the configuration in how you configure it.

It's important to recognize that all the existing sodium cooled fast reactors and most of the modern designs that are being developed utilize an intermediate sodium loop. What I mean by an intermediate sodium loop is that the sodium that goes through the reactor and removes the heat from the fuel, transfers its heat in what I've been calling intermediate heat exchangers to a secondary sodium coolant, and that secondary sodium coolant is then what goes out of the vessel in a pool reactor and goes to the steam generators in order to generate the steam which is then used to run the turbines for the electricity.

This was used in all the previous designs, and some of the benefits of this approach of having an extra loop with the secondary sodium is it isolates the primary loop from the steam generators, so you're not taking the same coolant that's going to be going through your reactor and having it go through the steam generators. This means there's no impact if you have failures in the steam generators, and that was a reliability issue in some of the earlier SFR experience.

This also allows you to keep everything near the reactor vessel and inside the reactor vessel at low pressure. Because you're operating with sodium very far from its boiling point, you do not need to pressurize the primary coolant and you don't really want to have high pressure systems and piping around the core itself, and so this keeps all of that outside the vessel and you only have the secondary sodium scene, this interface with the high pressure systems.

And almost all the designs out there, actually the steam generators reside outside the containment. So you have the containment and then you have…

I've lost my connection again. I'll be back in just a moment.

Okay, sorry about that delay. So I was talking about the intermediate loop and all the systems have the intermediate loop within them.

Next, I'm going to briefly review some of the typical operating conditions for the sodium cooled fast reactor. As I just talked about, you're at near atmospheric pressure. On the coolant temperature range and the outlet temperature of 500 to 550°C is really based on the structural materials, and this allow you to use conventional stainless steels, and this is pretty much the limit of their range, but because of the operating conditions and the thermophysical properties of sodium, this means that between that outlet and any boiling of the coolant, you're going to have about 350°C margin to any boiling of the coolant. This gives you very different behavior than light water reactor systems where you've pressurized in order to either limit or to be at the very edge of the coolant boiling behavior.

The fast reactors operate at significantly higher power densities than LWRs. This 300 to 500 kW/liter is about three to five times the power density of light water reactor systems.

In order to have the type of behavior you want from the neutron balance that I was talking about earlier, you want to have a very tight packing. You want your material that you're trying to capture the neutrons in is the uranium, and either to convert it into fissile material or to have the fissile material that's in the reactor to fission and to sustain the chain reaction, so you want a very high density of your fuel material. So the fuel pins in a fast reactor system are typically very much more tightly-packed than what you have in an LWR.

A picture here on the next page. This is a typical assembly. This is from FFTF, this picture actually, and you can see the fuel pins here, which are smaller than LWR fuel pins, are almost touching. And you use this wire wrap that's just wrapped around the pins, and this is the only thing then, that wire wrap, that keeps the pins from being touching, so it gives very small channels between the pins, and the good heat transfer properties of sodium, with the sodium now coming up out of the page, the sodium is able, even with these very small coolant channels, to be able to remove the heat at these high power densities and to give you effective heat removal.

You have then, typically, as you can see, you have this bundle of pins. This is the isometric view of that. And so the coolant comes in here at the bottom of the assembly, flows up through the pins, and then out of the assembly, and then it would go up to the heat exchangers after it comes out of the particular batch of fuel pins.

Now these pins are typically put inside a wrapper or a duct, as it's called, it's called different things internationally. This allows you to control what the flow rate is within the individual regions of the core, and this also has a structural function for the reactor system.

The fuel itself, there are two different fuels that have been used in the test and demonstration reactors that I showed. You have, most of the international experience is with oxide fuel form, and you have both with enriched uranium oxide and with a mixed oxide form, which is uranium plutonium oxide. Some key aspects of the oxide fuel are a very high melting temperature, it's a ceramic fuel form, but a fairly low thermal conductivity.

In the US experience, particularly in EBR-II, was with a metal alloy fuel. This is a uranium plutonium zirconium alloy. That fuel form, and their initial work on that fuel form was motivated by its high density. It does have a significantly lower melting temperature but it has a much higher thermal conductivity than the oxide fuel form.

This difference in melting temperature and thermal conductivity implies that I roughly have to go to the same amount of overpower between the two fuel forms in order to get to where I would have any type of fuel damage.

Now the design issues for these fuel forms are different. Both the metal alloy fuel and the oxide fuel have been utilized in the test and demonstration reactors internationally that I showed. Both of those fuel forms have been optimized in some sense in order that they've been designed to accommodate fuel swelling and to allow significantly higher burnups than we have for LWR fuels. Both the oxide and the metal alloy have been demonstrated up to 20% burnup with the current pin designs that are used in the reactors internationally.

For the metal allow fuel, the behavior typically is limited by chemical interactions between the fuel and between the cladding material itself. This tends to limit what the temperatures are that you can go to with the metal alloy fuel. There's a variety of advanced options being looked to overcome some of these, but that is typically the limitation on the burnup and on the temperature for the metal alloy fuel form.

Conversely, for the ceramic fuels, with the base being the oxide fuel, you're limited by the mechanical interactions. This is the typical thing that we worry about in LWRs, also where the fuel starts to swell and pushes on the cladding, and that's the behavior that typically limits you on the burnup for the oxide fuel.

There are also some differences in the fuel coolant compatibility. Metal alloy fuel is compatible with the sodium coolant, whereas the oxide fuel reacts chemically. This just means that on the oxide fuel you have to quickly detect fuel failures and you are able to then locate them and to make sure that you remove those from the reactor before the fuel interacts with the coolant.

This is a high level comparison then of the behaviors of LWR compared to sodium cooled fast reactor. These are some of the differences that I talked about as I went through the features of the fast reactor. You can see here, there's about a factor of 3 difference in power density with the power density being significantly higher in a sodium cooled fast reactor compared to an LWR. You have the differences in pressure with the sodium reactor at basically atmospheric pressure compared to the high pressure of a PWR system.

Outlet temperature on the sodium system is around 500°C, whereas you're around 330°C for a conventional light water reactor system.

Now the fuel form is a higher enrichment fuel. Instead of around 4% fissile, you're up around 20%, but it goes to a significantly higher burnup, and again, this is the behavior of conventional SFRs that are able to get up to about 10% enrichment as compared to 4% in current LWR fuels.

I don't have time to go into it in detail today, but I'll give a few quick words on the safety approach for the sodium cooled fast reactors.

The safety approach has two aspects that I want to emphasize here. The first one is a reliance on inherent feedbacks, and these inherent feedbacks are unique to the fast reactor system. There are some words here in the middle two bullets that describe what this is, but basically, the fact that you have high energy neutrons leaking out of the system makes the system sensitive to changes in the geometry. And if you design your reactor correctly, this can give you significant negative reactivity feedbacks.

Basically, how those reactivity feedbacks work is, when your system starts to heat up and starts to go over power and gets hotter, the materials expand, just by axial expansion of the materials, that expansion then allows more neutrons to leak out of the system, and that brings you a negative reactivity, which brings your power back down. So this inherent feedback is very important and it has been designed into and is a reliable inherent effect because you will have axial expansion as your temperatures start to increase. You will have thermal expansion, excuse me; it's not just axial expansion.

The other aspect of sodium systems is the coolant. The low pressure operation is important but you also have very good natural circulation behavior from sodium in the range you have. You have a large margin to boiling, but you're able to get significant natural circulation within that range, so all of the modern designs and especially all the Gen IV designs have passive decay heat removal systems which rely on this natural circulation behavior.

Now these safety features and aspects of fast reactors and sodium fast reactors have been demonstrated. In particular, the negative reactivity feedback was demonstrated by very severe accidents that were conducted, in particular in the EBR-II and FFTF reactors. In EBR-II you actually had an event that was done where you took the system and you stopped the pumps. You did not have the control rods insert with the usual SCRAM mechanism. The system started to heat up because the pumps weren't running anymore and it wasn't removing heat as effectively, and the system shut itself down because of these negative feedbacks and was able to then equilibrate at decay heat removal levels, with natural circulation then removing the decay heat.

So that's the basics of the safety approach.

The last topic I'm going to cover is the Generation IV international collaboration on SFRs.

We list here, and this is a review from the first webinar, so I'm not going to spend much time on it, but this is the goals for Generation IV. There are four areas that Gen IV was targeted in the goals it had for the behavior of what we would call Generation IV systems now, so Generation IV SFRs for this presentation: sustainability, economics, safety and reliability, and proliferation resistance and physical protection.

Some of the specific criteria among those four, the eight goals actually among those four broad areas are listed below, but those were the four areas in which we had particular targets.

Gen IV was established back in 2001. There was a Technology Roadmap activity conducted in the first two years there with the initial report coming out as shown there in 2002. That roadmap of advanced reactor options that could meet the Gen IV goals was updated recently, in January 2014.

As a result of that roadmap, there were six systems identified, and again, I'm going to highlight here the fast reactor systems. On the fast reactor systems that were identified for future work were the sodium cooled fast reactor, lead-cooled fast reactor, gas-cooled fast reactor, and there are also fast spectrum versions of the supercritical water and the molten salt option.

But the six reactor systems that were identified in that roadmap are shown on the lower half of this. They typically are talked about based on what their coolant technology is. The very high temperature is gas coolant, sodium coolant, supercritical water, gas, lead, and molten salt. In this particular talk, I'm going to be focusing on the SFR system, but you will be hearing about these other five systems that were identified in the roadmap in some of the subsequent webinars.

The Generation IV sodium cooled fast reactor, from the very highest levels, what are the missions? The mission here, and the big mission that you'd get very favorable performance from these systems, is the fuel cycle performance benefits, as I talked about in the first part of this talk. You're able to get improved utilization of uranium and waste management benefits.

We're also looking in the Gen IV collaboration at technology innovations that can reduce the cost of these systems, those innovations, and the cost reduction will make SFR also an attractive option for electricity production.

As I talked about, the operating outlet temperature range for these systems is 500 to 550C. There's a wide range of different power systems being looked at, from small reactors to very large reactors. I talked about the two different fuel forms. And these variations in the breeding ratio are basically to support the different fuel cycle options that I talked about earlier.

In the Generation IV, there's three different types of SFR systems that we're looking at. There's loop systems, there's pool systems, and there's small modular systems. What I show pictures of here are four design tracks. Design tracks are where one of the Gen IV members is contributing a particular Gen

IV design to be considered as one of the options for the Gen IV SFR. The four current design tracks are the Japan sodium fast reactor, which is a large 1,500 MWe loop system. The European sodium fast reactor, which is also a large same range, around 1,500 MWe pool reactor system, KALIMER, which is a 600 MWe system contributed by Korea, and then the small reactor design is about 100 MWe size contributed by the United States.

The seven members of the R&D collaboration on sodium cooled fast reactors are shown here. It's China, the European Union, France, Japan, Korea, Russia, and the United States. The work is organized for the Generation IV collaboration into five research and development projects. Those five projects are listed here. There's a project on system integration and assessment, there's a project on safety and operations, a project on advanced fuels, a project on component design and balance of plant, and lastly, a project on fuel cycle demonstration, which is Global Actinide Cycle International Demonstration. I'm just going to give very, very brief discussions on these particular projects to conclude this talk.

The members of the different projects are shown here. All seven of the members on the SFR collaboration are part of the system integration project and the advance fuels project and the safety and operations project. A subset of the group is currently involved in the component design and balance of plant, although there's discussions of the other members joining that. And then the GACID collaboration and the global demonstration is a three-country collaboration

One of the recent contributions from the advanced fuels project is a down selection report where they looked at different fuels and what were their recommendations for Generation IV systems, and this was completed in 2015. They recognized in that after several years of working together that the final selection on fuel type is dependent upon multiple factors. There's what experience each country has does definitely impact what types of fuel forms that they've targeted for the development of these future systems.

The final recommendations from that, actually, you had four of the seven countries, China, France, Japan, and Euratom, which are looking at oxide fuel for their Generation IV SFR. The United States and Korea are working on metal alloy fuel, and Russia is looking at nitride fuel as their advanced fuel option.

The GACID project is a good example of the collaborative R&D projects that can be conducted in the Generation IV context. In particular, what you have here is you have two countries, you have France and the US, working on the raw material preparation, and then the material is fabricated into fuel pins in France, and then those pins are irradiated in the Japanese fast demonstration reactors, the intent being to do this in Joyo and to then do the eventual assembly testing in the Monju reactor.

So you have the workload divided between the three countries, and then you had three phases for this project where you were first looking at neptunium and americium pins. Then you were extending the work to curium pins and eventually looking at bundles of pins. And again, the purpose of this project was to demonstrate the actinide recycle and the performance of that.

Just some of the topics for the component design and balance of plant project. You have work on in-service inspection and repair technology, both on the experience that was had in previous demonstration reactors and on the development of new instrumentation, and to new techniques for inspection.

Leak Before Break methodology has to do with the structural lifetime and the ability to predict that and the behavior when you do start to get failures. You also have for steam generators the development of new detection techniques and the ability to monitor the steam generators for both if there are any problems, let's say, of leaks or other aspects, as well as how close they are to the performance goals that you have.

But we also have within here the development of new energy generation technologies. There's been a very significant collaboration on the super critical CO_2 Brayton cycle, which would be a replacement energy conversion technology for the steam cycle in sodium cooled fast reactors.

Lastly is the safety and operations project. All seven countries are part of this project. We are looking at the analysis of experiments. There is some sharing of results. We're also looking at the development of computational tools, and we're looking at the features which are being included in the modern designs and some of the aspects of those, as well as the motivation for those being shared among the members. There is work also in this project on looking at what was done in previous reactors and then how that can apply to the safety and operations aspects of Generation IV system.

So to summarize then, the three topics I talked about, firstly, were that fast spectrum has favorable neutron balance and that favorable neutron balance is important and enables improved fuel cycle performance, both resource utilization and waste management.

The sodium cooled fast reactor is the most mature of the Generation IV technology options. I talked about the international demonstration experience. There are several collaborative Generation IV R&D projects which are working on improvements to that technology; in particular, they're looking at technology innovations that improve the economic performance and the robust performance and behavior in off-normal conditions with an emphasis on the inherent safety and those types of features within the Generation IV SFRs.

And that's the conclusion. Thank you very much for your time.

Oates: Thank you, Dr. Hill. If you have questions on the presentation today, please do type those into your chat box. I only see the one to define IRT on slide 10 at this time. I'm going to mute my line for a minute. I've got an alarm going off in the building. Hang on.

Hill: Yes, IRT on page 10, yes, there were lots of acronyms in the original version of that. Sorry I didn't capture that one. That was the independent review team. The point there was this was conducted by a large team on this fuel cycle study and the results were reviewed by an independent review team with a broad set of experts with both nuclear background, as well as economics and other backgrounds.

Yes, there was a question on the cost of electricity in Russia for BN-600. That's a very hard question to answer. The way they do the cost and the charging of it is different in Russia compared to other countries. And you have to remember, BN-600 was a demonstration reactor. It was not the Nth of a kind generated that. So I don't have a specific number on what the cost is. I can tell you that when they did the BN-800 design, the BN-800 produces 880 MWe compared to 600. It's almost exactly the same size as the BN-600 system, so they're expecting to get a significant improvement in the capital cost behavior on BN-800 compared to 600, but I cannot answer the question quantitatively on what the cost of electricity is.

The question on, "With the current state of knowledge on SFR, would you say SFR technology is right for commercialization?"

SFR technology is mature enough that you can look at commercial demonstrations now, and that's what you have being looked at in most of the international groups. You have the JSFR, the Japanese design. The Russians are working on a BN-1200. You have a couple of different US vendors talking. Most of those designs are what I would call commercial demonstrations. The intent would be to build a first one of those and then you would replicate that, and, yes, that's there. What you would like to do is include some of these features on the Generation IV systems that we've been doing research on because those will help to bring the costs down of the system, as well as to make the safety performance even more robust.

Okay, a good question here. "Why is the fact that sodium cooled fast reactors have a positive void coefficient not a bad design feature for them as they are for LWRs?"

The reason why is it's different. You can't look at it like you look at it in an LWR. In an LWR, the reason the coolant isn't voiding is because your system is pressurized. Basically, if you have anything happen within the LWR that you lose pressure on the system, you get much more voiding of the coolant. Conversely, on the sodium reactor, you're operating with an outlet temperature that is 350° away from the boiling point of the coolant, and in order to get over that 300°, a lot of things have to happen in the reactor system, and it gives you time for these inherent feedbacks that I was talking about to bring your power back down to where you don't have the temperature continuing to increase.

So what you have for the sodium systems is you have the ability to go through very severe accident conditions and transient conditions and still have significant margin to any boiling of the coolant, which makes it a very different issue. We still do look for these designs on what the behavior is with the voiding for very, very severe conditions, but it is not the same type of issue that it is for water-reactor systems or other systems that are using a coolant that's close to its boiling point.

There's a question here about, "Any advantage of SFR beyond fuel cycle? Can you say more about AFR?"

Yes. As I talked about when I showed that picture of the fuel cycle, there are benefits of fast reactor systems for small reactor applications, because of you have the high-power density of the system, your small reactors can be smaller than they would be for thermal spectrum systems, and that is being looked at. The design I showed on the 100 MWe, AFR 100, is an NSFR version of the small reactor designs, and that's one aspect that's being looked at, and you definitely want to look at them for electricity production.

The other aspect is for instance there has been a proposal talked about in the United Kingdom where they're looking at plutonium management, which is also sort of a fuel cycle issue but a bit of a different fuel cycle issue than just closure of the fuel cycle.

I'm trying to go through these questions, so I apologize if I haven't gotten to yours yet.

A question, "Is anybody currently building a small modular AFR 100 reactor?"

No is the answer, and the cost estimates are very immature. This is a preconceptual design at the moment.

A question about the C4 site and the nickel panels used for control in that with gravity drop, SCRAM system achieved for coolant loop systems.

All of the modern SFR designs use control systems and SCRAM systems that in many ways look similar to what you have in LWRs. Typically, these are bundles of rods, and those rods contain for most designs boron carbide, and they basically can either be driven into the reactor system or can fall by gravity into the reactor system from above the reactor and then to cover the active height of the core region.

Ultimate disposal of the sodium.

We have had experience with that. The EBR-II reactor has been closed down now and they have basically removed all of the sodium from that vessel and they have basically what you can do is, the sodium, eventually it gets burned and it's not radioactive because the radioactive species that you do generate by the slight activation is very short-lived. You're able to clean that up and it has been done with the decontamination and decommissioning of the reactors. I would have to go into a lot more detail for some of the precise aspects of that, but there is experience with that.

I was going to keep going through. Is it okay to cover some more questions here?

Oates: Absolutely. I've lost track of where you're at is all. Can you say the name of where you're at?

Hill: Yes, I'm on Jan Reynold Agustin's question.

Oates: Got it.

Hill: On, "Reliability/safety issues with water and sodium coolant mixing?"

Yes, this is something that has been looked at from the day you started working on sodium technology. The key aspects on this is that, as I talked about, all of the modern designs and all of the existing demonstration designs use an intermediate loop, so if there are any issues with the sodium interacting with water, that's not affecting the primary coolant so it's not an issue of the nuclear behavior of the system or nuclear concerns on the safety behavior.

It is an issue because you don't want to have problems with your steam generator, which can vastly impact your reliability of your system, and indeed, in the early demonstrations, this was some of the problems that they had. PFR and BN-350 both had some problems with leaks in their steam generators. They had engineered systems to deal with those leaks, but they did cause some reliability issues in those early reactors.

If you look at the experience though over the last 35 years from BN-600 operating at 80% capacity factor, and I think those engineering issues have been dealt with to have safe steam generators and reliable steam generators that do it, but you still are doing it in a manner where it's with the intermediate loop sodium, not with the primary sodium.

Okay, we have a question here on the estimate of the difference in capital cost between SFR and traditional LWRs.

The costs on the demonstration reactors that I talked about have been higher than LWRs. That's typical what you will have for a demonstration reactor. You are fairly conservative on how you do the design and how you operate it.

And I lost my connection now. I'll be back in a moment.

Okay, back on. So the demonstration plants that you've had to date obviously will have higher costs than LWRs. The Generation IV designs, there have been some fairly detailed cost estimates and both the Japan sodium cooled fast reactor and the design the Russians are working on, BN-1200, those designs have cost estimates that indicate lower cost than the comparable LWR systems, not vastly lower costs but lower costs than the comparable LWR systems. Now those have not been built, so you don't have proof of that principle yet, but you do have indications that some of the designs that include the modern features can get to capital costs lower than LWRs.

A question on the role proliferation is playing in the selection of current designs and possible timetables to their implementation.

There has been a concern with proliferation issues traditionally. This has been one of the big debates on the implementation or not of closed fuel cycles. Most of those issues with the closed fuel cycle and the concern with any proliferation issues actually don't reside in the reactor; they reside in the remainder of the fuel cycle. In the reactor, you're handling a bundle of pins and you're putting it in and you're tracking it and then you're taking them out, and there's not a direct concern with those pins. It's when you're later doing the separations on the material that usually you have the concern, but it has definitely been an impact on whether you do or you do not both prioritize and the rate at which you implement this technology.

Now this has been one of the big focuses of the Generation IV collaboration, and the PRPP group, which is probably going to give a seminar eventually in this webinar series, has been looking at ways and aspects to basically improve any of the physical protection on the fuel cycle for these future reactors, as well as looking at features that have more intrinsic proliferation resistance. So that is being looked at and you do have some options today you didn't have in the past, but that's a continuing debate on if there are issues for the closed fuel cycles or not.

Okay, let me move down here.

Okay, overall waste of the sodium cooled fast reactor compared to operating LWRs, as well as balance of high level waste to low level waste.

At the very highest level, the simplest level, you're going to have fission products that need geologic and long-term disposal, and those are going to be generated, the fission products, at the rate that's proportional to the amount of power you produced in the reactor system. So it's not by doing these advanced fuel cycles that you've eliminated the need for any long-term waste management; you still will have it.

What you've done with the recycle options with the fast reactors and the ??? management is you've excluded the actinides from the waste, so the uranium and all the transuranic species you can exclude from the waste. That's important for several aspects of the waste, on the high-level waste in particular. It gives you significantly less mass because the uranium is a very significant portion of the fuel that's going to the waste, and it gives you significantly less long-term radiotoxicity and long-term heat compared to if the actinides are in there. But in the short-term, things like heat load are dominated by the fission products which will be in the waste from either type of system.

On the low-level waste, I don't expect large distinctions. Those are typically associated with running a reactor. I don't know that there's any particularly large differences that have been identified between different nuclear systems.

The four types of Generation IV reactors, which one do you think will be the most successful and why?

I'm going to have to not answer that question because the fact that you have four different countries contributing four different tracks, there's a definite difference of opinions on what the market's going to be and whether you're going to want to go small modular, medium-sized modular, or large monolithic plants. So I can't answer that question.

"How often would outages need to occur, and how does the decrease in power during an outage affect the sodium coolant?

Most of these reactors are designed for similar cycle lengths as we're running LWRs today. The designs that are operating at fairly high power density are typically running somewhere between 18 month and two-year cycles. So you would come down for a partial refueling on that rotation, and then you would be down and you'd replace part of the core and be back up and running, hopefully in a very short time outage, and they have shown short time outages.

When I said they have shown that, BN-600 actually operates on only a six-month cycle but they've managed to operate in an 80% capacity factor over a 30-year period. So the decrease in power doesn't really have large effects on the sodium coolants. There are some activation products in the sodium coolant that decay away fairly quickly. You do have a very short wait that you would want to do on that, but you keep the system inerted anyways, so you're not directly having the sodium see the environment, actually.

And what you do have to make sure for these liquid metals is that you do have some level of heat. Usually the decay heat once the reactor is up and running is enough to keep the sodium melted within the large pool, but you do have to have systems and heater systems in early times for the reactor in order to make sure you don’t freeze the coolant.

That system, for sodium coolant that's not a huge challenge. Again, the decay heat and even running the pumps usually can give you enough to keep the sodium molten. That becomes a bigger issue for some of the other technologies that have a higher melting temperature.

Advantages of the supercritical CO_2 cycle over steam cycle with respect to safety.

First of all, you do not get a significant difference in thermal efficiency at SFR temperatures between supercritical CO_2 and well-configured super-heated steam cycles, so they are roughly the same efficiency. What you do hope and what you have shown, if you can get the supercritical CO_2 technology matured, is that they will be much smaller. The turbines for the supercritical CO_2 are much smaller than a steam cycle, so you have hopes that the energy conversion cycle equipment will be less expensive than it is for steam cycle, but it's not going to be an efficiency benefit at these temperatures.

Now with safety issues, again, it's an industrial safety issue, not a nuclear safety issue. You will not need some of the systems that you currently have in these reactors to account for the fact if I would get leaks in my steam generator and have to deal with the pressure pulses that would result from that. That's an issue for reliability and for keeping the steam generators and the energy conversion system running, but because of the intermediate loop, it's not a nuclear safety issue.

How are leaks and breaks treated in case of failure in the detection system?

You have to have, as you've noted in your question, a good detection system. That's actually fairly easy to do with the sodium system because any leaks that are not in an inerted area are going to give you lots of smoke, so you quickly detect those. Typically what you do when you get those is you will freeze the sodium in that area. You will remove that portion of the piping, and you will replace it. This has been done during reactor operations. They've done this several times over the years. You can look at the BN-600 documents on their history on doing that.

You need to have obviously multiple ways to do the detection, and obviously what's the failure, that's going to be the exact question on the detection system, is to try to make sure eventually you're able to catch it and quickly capture and replace the particular part that has the leak.

Yes. There's a question on the fuel cycle options study – did it find proliferation to be a non-differentiating factor for most/all of the evaluation groups?

That is true. Part of that aspect, and what I was trying to address in the previous question was, what are the issues that people raise for these closed fuel cycles? The advantage on the closed fuel cycles is when you have the closed fuel cycle, and especially when you have the resource extension version of the closed fuel cycle, you don't have to have enrichment. And so then part of this whole proliferation evaluation is what are the risks associated and the concerns with enrichment technology versus the recycle technology, and I'm not the expert to comment on that. Please save those questions for whoever from the PRPP group is going to be giving a webinar.

A question on the severe accident characteristics of oxide fuel with metal alloy fuels.

We have observed the ability to prevent coolant boiling more readily with the metal alloy fuels compared to the oxide fuels, and that's because of the difference on thermal conductivity between the fuel forms. The fact that the oxide operates at lower temperatures gives you the ability to be more resistant to driving the sodium temperature up to its boiling point, even in very severe conditions. So the prevention aspects, there is some benefit there.

As far as once you get to the point where you're getting fuel failures, it's very different behavior between the oxide and the metal alloy fuels, and that's being looked at on the differences in behavior. Again, because of the difference in melting point, there's very different behavior between the two fuel types.

Can the SFR be designed to load follow?

Yes. There's some benefits for fast reactors for load following because the same negative feedbacks that give you the good safety behavior make the system automatically tend to adjust. In other words, if you're pulling from it a different power and you're trying to pull more power or less power from it, and you're doing that in your intermediate system, the system will tend to adjust to the power level that you have, and you don't have as severe of thermal shock issues with the fast reactors as you have largely because of the high conductivity of the sodium and you're in a pool of sodium in most of these designs.

Yes, Alfredo is absolutely right, that on the waste issue there is some distinction between the fast reactors and the light water reactors because fast reactors operate at a better thermal efficiency. What that means, they typically can operate at around 40% thermal efficiency as compared to 33% for LWRs. That means for the same amount of electricity generation you would have less waste for the SFR compared to the LWR.

Gas conversion system in SFRs replacing steam generators?

We were having a lot of discussion of this in Gen IV now because the ASTRID design is talking about using a nitrogen energy conversion system. Supercritical CO_2 was largely targeted in order to avoid any concern or having to look at sodium-water reactions. The biggest issue with some of the gas conversion systems, like the nitrogen that's being looked at for the ASTRID design, is there's lower efficiency for those systems in this temperature range compared to steam or supercritical CO_2.

There's a question here on the deterioration of the sodium coolant. Do they need makeup of the coolant fluid?

We have not seen any need for makeup of the coolant fluid. There is a well-demonstrated technology for coolant purification for sodium, so they are always running with that purification system running. Basically, for sodium, the purification system is to keep the oxide content in the sodium very low. That's done through cold trapping, which only requires you to divert a small fraction of the flow in order to do the coolant purification.

Once you do that, you get very little corrosion with the types of structural materials that are being used and you've operated these reactors 30 years for EBR-II, going on 36 now for BN-600 with no need for coolant makeup or for any pulling the coolant out and cleaning it up beyond what's done in normal operations.

A comment on the apparent favored use of oxide fuels internationally over US preference for metal fuels?

The main source of that difference is experience. Internationally, there is a lot of experience with oxide fuels, for instance, on the French side, Phénix used mixed oxide fuel, Superphénix was designed with oxide fuel. That's their experience base; that's their infrastructure. In the United States, EBR-II operated for 30 years with metal alloy fuel. FFTF operated for a shorter period with oxide fuel, so there's more experience with the metal alloy fuel in the US. So I think a lot of the difference on preferences you see is based on experience.

You can contact me to learn more about the AFR-100. I can get some references to you.

I think I've gotten through the list of questions.

Oates: I think so as well. Thank you so much. There is so much interest evidenced by the amount of questions. I apologize for having that alarm go off. I had to step away so I didn't get the earlier ones posted.

You can see on your screen the upcoming presentations. In January, there's a presentation from Mr. Carl Sink with DOE on Very High Temperature Reactors. That will be followed by a presentation on Gas Cooled Fast Reactors by Dr. Alfredo Vasile. And in March, a Supercritical Water Reactor, a presentation by Dr. Laurence Leung, with Canadian National Laboratories.

Thank you again, Dr. Hill. I know it takes a bit of effort to put these together, and your energy and effort is well recognized.

Hill: Thank you.