Supercritical Water Cooled Reactors
Dr. Laurence Leung, CNL, Canada

Berta Oates: Good morning, everyone, to the next Gen IV International Forum webinar. This presentation this morning is on supercritical water cooled reactors. The sound is broadcast over your computer speakers. There is the Q&A pod, where you can type in your questions for today's presenter, and we will take those at the end. Again, the PDF version of the slide deck is in the Files pod. You click on that and it will download to your computer. And lastly, but probably most importantly, is the survey for your feedback, which helps us improve the webinars and gives us information on your thoughts and what you would like to see in the future as well.

Doing today's introduction is Patricia Paviet. Patricia is the Director of the Office of Materials and Chemical Technologies, Office of Nuclear Energy, and she is also the Chair of the GIF Education and Training Task Force. Patricia.

Patricia Paviet: Thank you so much, Berta. Good morning, everyone. It's my privilege to introduce Dr. Laurence Leung from Canadian Nuclear Laboratories today. Dr. Leung has been working at CNL, formerly Chalk River Laboratories, of Atomic Energy of Canada Limited, since 1987 in the field of thermal-hydraulics.

He completed his PhD degree at University of Ottawa, Canada, in 1994. He is currently Manager of Research and Development Facilities Operations, and he is also responsible for the development of the Canadian Supercritical Water-Cooled Reactor concept.

He received 13 awards from Atomic Energy of Canada Limited, AECL, an external organizations, and delivered short courses on thermal-hydraulics and supercritical water cooled reactors.

He is one of Canada’s representatives to the GIF Supercritical Water-Cooled Reactor System and is Co-Chair of the System Steering Committee and the Thermal-hydraulics and Safety Project Management Board.

So it's my pleasure to have you here, Dr. Leung, and without any delay, I give you the floor.

Laurence Leung: Thank you very much, Patricia. Good morning, ladies and gentlemen. Welcome to the webinar. I am providing this presentation on behalf of the SCWR System Steering Committee members, so let's get going.

Okay, this is an outline of my presentation or talk. First, I will give you a brief historical development. Since the 1950s, we started the concept development already. And then give you some system designs and materials and fuels, and the ??? configuration that we have right now for different concepts. And mention about some specific application other than power generation, the status of the concept development at this point, and we have several partners participating, some of us are completing the concept already and others are still working on their concept.

And one of the areas that we are interested in is how the SCWR concept aligns with the GIF technology goals technological, so there are a couple of the technologies set out by the Generation IV International Forum for developing a new nuclear system, so we see how the SCWR matches to those technical goals.

I will mention some of the challenges that are facing right now, and some of them we have resolved, and others we are still working on. And the majority of our work is through the collaboration ??? or others in their national organization, so I will briefly talk about that.

At the end, I'll give you a quick summary.

So, I think the first question that we always ask is why a supercritical water cooled reactor? We are looking at the landscape right now. There's a lot of advancement in technology between nuclear and fossil fuel power plants, and the SCWR actually is merging the two advanced technologies together.

If you look at the system, we are basically using the core for the nuclear system and the balance of plant for the ??? fossil-fuel power plant to come up with the SCWR concept. So this is a way that we will only focus on the core area and we don't have to worry about the balance of plant because we can simply take the balance supply concept for the advanced fossil fuel power plant.

And if you look at the landscape, the majority of utilities providing electricity or power, it actually operates both nuclear and supercritical fossil power plants, so it's much easier for them or the utility to adopt the technology. And because we have been working on both the nuclear and fossil fuel power plant for the past 50 years, we have many years of design and operating experience, so that will help us as a strong background to help us move forward with the next-generation of water cooled reactors.

So the SCWR main features, the main feature for the SCWR is high efficiency because we are operating at supercritical pressure and temperature at the core outlet. And you can see on the right-hand side is a figure comparing the pressure and temperature with a different system. You have the BWR and 7 MPa, the CANDU PHWR, pressurized heavy water reactor, about 10 MPa, and the pressurized water reactor at 15 MPa, and the SCWR is operating at 25 MPa.

The unit temperature is a very low range between the pressure vessel type and the pressure tube type, so we will go over the different concepts and operating conditions later on.

So the increase of high efficiency actually has several advantages because you can use the same amount of fuel but generating much more power, so the power output is higher than the other concept. And then, because of high efficiency, you can reduce waste from the turbine and the condenser, and that will help the environment. And because you need less fuel, you can fuel fewer plants to meet the demand, and that will help us to save capital and operating costs.

And the majority of the concept actually applies a simplification of the plan because we use a direct cycle, so the direct cycle allows us to pass the steam directly from the core to the turbine and means eliminating any heat exchanges, steam generators or steam dryers or moisture separator, reheater in the other system. So that will help us to reduce capital and operating costs as well.

So the development of the SCWR actually is very flexible. We can either have a thermal spectrum or fast spectrum concept, and we can implement an advanced fuel cycle, for example the Canadian SCWR implements the thorium fuel cycle, while other systems using the uranium fuel cycle. The fast reactor actually, the fast spectrum reactor actually implements the MOX fuel cycle.

And then, because definitely we reduce the electricity generating cost and because of the high temperature at the outlets, we can also have the opportunity for co-generation as well, and those are the things that I will talk about later on.

So historically, there's been a proposed SWCR concept since 1950 and 1960, and they have the Westinghouse Supercritical Reactor concept in the thermal spectrum with a 70 MWth operating at the pressure of 27.6 MPa. The other one is a supercritical once-through tube reactor, SCOTT-R, and it's a big reactor operating with 2300 MWth and at a pressure of 24 MPa. And we also have the General Electric Hanford supercritical reactor. We have 300 MWth at 37.9 MPa. The Babcock & Wilcox supercritical fast breeder reactor, proposed 2326 MWth, for operating at a pressure of 25.3 MPa.

So all of those are concepts. We don't have any supercritical water cooled reactors you are operating yet, but we do have a few superheated steam reactors at high temperature but lower pressure. For example, in the Russian Federation, the Beloyarsk AMB-100 and AMB-200 reactors, they operate at 510° superheated steam. And then the Heissdampfreaktor at 457°C in Germany. So we do have some experience in operating the reactor at high temperature, but not yet at the supercritical pressure.

So since 1960, the development of the SCWR concept has been dormant for quite a few years, and in the 1990, it started to have renewed interest, mainly because of environmental concerns and trying to reduce greenhouse gas emissions in the global environment. And also, we have a demand for stable energy and supply, especially for the developing countries, and because of the development in the supercritical turbine, there's the potential for cost reduction. Usually, for nuclear, the fuel cost is lower but the capital cost is higher than the coal-fired power plants, so by increasing the steam temperature we can simplify the nuclear system. For example, we can use the direct cycle so that we can move the steam or pass the steam directly to the turbine.

And the advancement in the boiler technology, you can leverage the development in the fossil power industry reducing the cost and risk because we can simply adopt the balance of plant for the fossil power plant, and there we save as a development cost.

And in the fossil power plant, they actually can go up to achieving almost 700° in steam temperature and achieve 50% efficiency. So we still are not yet there, but we are approaching those high-temperature operations now.

So the supercritical water cooled reactor concept development, we have several partners jointly participating in developing the concept, and Canada, China, the European Union, Japan, and the Russian Federation. All five partners have signed the GIF SCWR System Arrangement to jointly develop the concept. I mean, all the concepts are actually evolved from the current fleet of nuclear reactors. You see the pressure vessel type which is evolved from the boiling water reactor or the pressurized water reactor, so with similar, you have a pressure vessel with the core and the fuel is inside, and then the steam directly goes to the turbine. And then you also have the pressure tube type. The pressure tube type is, for the core side, it's quite similar to the other concept except that the core itself is separated into different fuel channels, and the steam is also gathered at the outlet header and then passed to the turbine directly.

So you can see that even within the core, there is a lot of similarity between the pressure-vessel type and the pressure tube type, and based on that, we see a lot of commonalities that we can have, the research and development working together to advance the technology.

So I'm going to give you a number of core concepts that we have developed so far. As I mentioned before, the flexibility allows us to have a thermal spectrum or other spectrum, a fast spectrum, and we even have a mixed special concept.

So for the thermal spectrum, you see, here's the Canada pressure tube type, the SCWR core concept, and the core itself is separated into the top part, this is the inner plenum area, and then the flow is actually coming in from the liquid pipes, the water pipe, and gets into the core area and gets into the fuel channel, and then flows down in the middle pipe. I will show you the details later on. And then you take a reverse and pass through the fuel, and then go into the outer plenum in this area, and then go directly through the steam pipe to the turbine.

The core itself is separated into, like I said, the inner plenum, and then you have a low-pressure calandria similar to the pressurized heavy water reactor, so the moderator is a heavy water separated from the coolant, the coolant is light water. And the rod, this is the control rod, and the shutdown rod, it either comes from, right now it's a (reference) and it comes from the side or it can either come from the bottom as well.

And the second one is the China pressure-vessel type, and the core is similar to the PWR. The core itself, this is a fuel area, and then it's separated into the top part, the liquid coming in, the water coming in, and then going through these two parts, one part is getting done and then come back in the second part, and then go out the to the steam pipe to the turbine. The control rod and the shutdown rod come up from the top part and then are introduced into the core.

And the European Union, the pressure-vessel type SCWR core concept is similar to the China pressure-vessel type. It's also similar to the PWR, and the difference is this concept has a three-part system, so the water actually comes in from the top, going through the evaporator shown, and then it will go down to the bottom, come up from the evaporator zone in the core, and then get down forward to the super heater zone 1, and then mix at the bottom and the come out again through the super heated zone 2, and then get out on the steam pipe to the turbine.

The other thermal spectrum is Japan's pressure-vessel type, SCWR. It's also a single part system. It's similar to the PWR, and you can see that the core itself is in this area, and before the fuel is introduced, in the middle, and the flow, actually the inner water comes in, and then separate, the top part is to cool down the top part of the vessel, and the majority of the flow actually comes down to the bottom, and then mixed at the rod and then comes up through the core, and then goes out either to the steam pipe and then to the turbine.

So the control rod and the shutdown rod actually come up from the bottom similar to the PWR configuration.

So the other type or the Spectra spectrum concept in China, and Shanghai Jiao Tong University actually developed a mixed core, so this is a separate into two zones. This is a thermal zone and then the flow actually comes up and then passes through the thermal zone in the downward flow, mixes at the bottom, and then goes up through the fast zone of the reactor, and then gets out from the steam pipe that has the turbine.

And the second part is Japan is the fast-spectrum SCWR, and then it's similar to the PWR concept. The core itself is in this area and it's also separates in the blanket zone and the seed fuel, and then I'll show you the core configuration later on. And then the flow is actually similar. This is a two-part system. So the one part is going through the blanket and then the other part goes to the seed area.

The Russian Federation is similar to the pressure vessel type, and then this is the core area, and it also is two different concepts. One is a single pass and the other one is a two-part system as well, and they are still developing the concept. So similarly, you have the water coming in from the core top and then passing through. It is a single pass, and it comes down and then passes through the fuel in a single pass and gets out of the core from the steam pipe. If it is a two-part system, then some part of the flow will drop down, passing through the first part in a downward flow and then in the second part come up through the second pass and then go to the steam pipe. So all the core that you see, the control rod, actually comes from the top and then goes through the core region.

So here is the diagram for the core map. As I mentioned, the Canadian SCWR actually is a single-pass system, so you have a three-batch refueling concept. That's the core map that shows you the refueling pattern, so the fuel coming in and the first burn in this area, and then after about one year, it's taken it out and moved to the second area, and then burn for another year, and then taken out and moved to the third area, a third zone, and then after the third burn, they remove it and then put it into storage.

China's thermal spectrum, the core map, it is separated into two zones. This is zone 1, and this is the region that we have the downward flow, and passing through the core region, and then after that, the mixing at the bottom of the core, and then coming up with the second zone, and this is the outer zone with the second flow coming up.

And the EU, the core is separated into three zones, so in the second, the middle one is called the evaporator zone, and that's the one with an upward flow. When the flow comes down to the bottom, it travels to the top and through this area. And then the second is the red area, which is the superheater 1 zone, and then this is the flow traveling downward and through the superheater zone 1, and then mixed at the bottom of the core and then coming back up with the superheater zone 2, the yellow rich area, and then going out the core.

The Japan thermal spectrum, again, this is a single-pass system. It shows you that they adopt also the three-batch refueling system, so we'll show you the first batch, the second batch, and the third batch refueling, and afterwards, the fuel is moved to storage again.

The China mixed-spectrum SCWR, so the thermal zone is out of region, so the flow is moving downward through the zone 1, the thermal spectrum region, and mixed at the bottom, and then coming up through the fast spectrum region.

The Japan fast spectrum, this is a color map, and the blue area is the seed assembly, and then the gray area, the light-gray area, is the blanket assembly, and the dark grey area is a reflector, so the water actually comes in from the top and then goes down through the blanket assembly and then comes up through the seed assembly, in this area.

The Russian Federation, they also have the fast spectrum core, and the gray area is the one with the control rod. The gray zone is the assembly with the control rod. And then the white area is the assembly without the control rod inserted. So you can see that similarity between those two.

So for the fuel concept, for the thermal spectrum, Canada's pressure tube type, the fuel itself is actually that the whole assembly in here is the fuel assembly, so the water actually comes in from the top, goes through the nozzle acting as the orifice to control the flow rate going down to the channel, and then passes through a fuel nozzle coming down through the center, the water paper in this area, and then reverses the direction at the bottom of the fuel assembly, and the moves up through the fuel rod itself.

And then the fuel rod has two rings. It's 32 rods in each ring, and the rod is separated by the wire-wrap spacer in this case. So it's a (feature into itself). It's only the pressure tube in this area, so after one batch, the fuel assembly will be taken out and then moved to another position. And then after the third batch, the fuel will be completely taken out and then moved to the storage area. So the pressure tube itself doesn't move and just stays in the core for, the estimate is around 75 years that it can stay in the core.

For China's pressure vessel type, one of the differences between this concept and the other is they are usually putting a hole in the center of the fuel palette, and then that will help to lower the center line temperature. And then the fuel is separated into two rings in this area, so outside is 9 by 9 rods, and this one, the diameter of the fuel is around 9.5 mm, similar to the PWR fuel, and then they also have a center, there's a water box over in ???, it's a 5 by 5 area. And four assemblies are grouped into a cluster, and then inside you have a crucible-type of control rod in between the fuel assembly.

And the European Union pressure vessel type fuel concept, the concept itself is similar to what they have in the other pressure vessel type. You can see this is a water box in between, and then this is, the rod, the assembly itself is 7 by 7, and then with the center, the water box is 3 by 3, so this is one assembly, and then they also use the wire wrap spacer in between the fuel rods. And then, in the center you have a water box and then it separates the fuel from the water.

The reason why they introduced the water box in here is to increase the moderation to the fuel in this area and that will help with fuel utilization. The fuel cluster actually contains nine of these assemblies grouped together, and then they have an orifice in place at the bottom or the top to adjust the flow rate through the whole cluster.

The Japan thermal spectrum, the pressure vessel type SWR fuel is a quite large fuel assembly. It's a 16 by 16 rod assembly in this area, and they also contain a water box, and the fuel itself is a graded uranium configuration at the different location. The bottom in here is to show you that this is actually graded at different locations. And then the fuel assembly is around 5.8 meters. They also suggest they have two different types, either the grid spacer or the wire wrap at this point that you are trying to select.

So from the mixed spectrum, they separate the reactor into two zones, so of course the fuel assembly will be also covering both the thermal zone and fast zone. So for the thermal zone, this is the fuel assembly. They also have a single rod assembly in the outside and then two rods on the inside. So the fuel assembly also contains a water box in this area to help in moderation, and they also apply graded uranium along the length so reaching from 7 at the lower end to 6 and 5% enrichment to the top.

And for the fast zone, there is no, they don't need moderation so the fuel itself is compacted into this area and the fuel itself, they are using a wafer type of configuration, so basically mix the blanket and the sheet area in the wafer form, so along the fuel rod.

So in the Japan fast spectrum, they separate into the seed assembly and the blanket assembly. So the seed assembly, this is the area in a hexagonal type of configuration, and the right area is the fuel rod, the dark, the black area is the control rod guide tube, and the blanket area, the fuel itself, the fuel rod that (comprises that), and then outside, you have a zirconium hydride layer as the moderator in the blanket area.

So the Russian Federation fast spectrum also has a hexagonal type of fuel rod configuration, and the outside, those are the fuel rods (revolve) the control rod, and then the dark black circle is the one with the control rod in place, and the fuel itself is also configured into like a cluster in this area.

From the safety system, the majority of the concept is actually very close to the ABWR, or next is ABWR as an ESBWR type, and it is all surrounded by a lot of water, for example, the Canada SCWR safety system, this is the core area, and this is the steam pipe connected to the turbine, and this is a gravity-driven suppression pool in this area. And then you have a reserve water pool at the top. The slight difference from the Canada SCWR concept is the safety system is actually equipped with a separate passive moderator cooling system because the concept itself is to separate the moderator from the coolant, so a separate cooling system for the moderator is introduced to allow the heat to pass through from the fuel to the moderator, and then the moderator to the reserve water pool. And then, the ultimate heat sink is actually air cooled. So there's a heat exchanger outside of the reactor building, and then the reserve water pool, the heat is transferred to the heat exchanger, and the ultimate heat sink is the air, ambient air, so as long as you have air, you continue cooling the reserve water pool.

And then, China's safety system is similar to the ABWR, so they have also the reserve water pool, and the core is over here. And they have the automatically depressurizing system, and then also the (ECC) comes in as well, the reserve water pool, and with the containment cooling system as well.

And the EU's safety system is similar, and then all of them are similar actually that the core is over here, the gravity-driven suppression pool on the side, and then those are the steam pipe and then the water pipe, and then connected to the reserve water pool.

And this is the Japan safety system. The safety system itself for Japan's concept is the same for both the fast spectrum or the thermal spectrum, so in here, this is the core area with the connection to this steam pipe and the liquid pipe. And those are the suppression pool. And also the reserve water pool is on top, and automatically a depressurizing system, and they are connected into the outer system, and this is connected to also the balance of plant and the cooling system.

And again, the Russian Federation's SCWR safety system, it's two areas that are focusing on… This is a suppression pool, and then the reserve water pool. This is the core area. And they have also the containment cooling system to reduce the pressure just in case of an accident scenario. But all of them are basically similar to, like I said, the ABWR, the ESBWR configuration.

So from the plan concept, Canada, the EU, and Japan have completed their concepts, and China and the Russian Federation are still working on their concepts. So they are similar. You have the core, the containment, the reactor building, and then connecting to the balance of plant area. You have the high pressure and the medium pressure turbine and the low pressure turbine, and the connected to the generator.

So slightly different is, in here, the Canadian concept, the high pressure and the medium pressure turbines also can be contained inside a containment in this area. And the rest of them are similar to the PWR.

The EU concept, this is the same. You have the core, the ??? for the steam pipe to the high pressure turbine, in the medium-pressure turbine, and the low-pressure turbine. And similarly, for Japan's concept on the plan, and this is the core area connected to the turbine area.

So I give you this as a summary of the key SCWR parameters. So you see that Canada is the only pressure tube type and the rest of them, China, EU, Japan, and Russian Federation, are all pressure vessel type. And Canada is thermal, China has thermal and mixed, the EU are both thermal, Japan has thermal and fast, and Russian Federation has fast.

The pressure itself is more or less the same, except the Russian Federation is slightly lower pressure.

Inlet temperature at the coolant, the majority are around 280 and 290. The Canada concept raises it to 350° inlet temperature. The outlet temperature, Canada operates at 625°, it's matching the advanced high-pressure turbine development in the fossil power plant. The rest of them are around 500 or 510, and the highest is 560 for the Japan thermal spectrum concept. The Russian Federation is also at the high end, 540.

So the thermal power that goes to them varies from 2300 up to 4000, 3800 MW. Thermal, except for Japan, the fast reactor, which is smaller at 1600 MW.

The efficiency, because of the high temperature, outlet temperature, the Canada concept has a higher efficiency at around 48, and the rest of them are around 43 to 46% thermal efficiency.

And the fuel itself, the majority use UO2, except for the fast zone that uses MOX fuel. Canada uses plutonium thorium as a reference fuel, but we also use UO2 or MOX fuel as well as fuel.

The moderator, most of them have light water as a moderator. The Canada thermal core spectrum concept is using heavy water, and the Japan fast blanket size uses zirconium hydride as a moderator.

The flow pass, there is some difference between each concept. Canada uses a single-pass system and China uses a two-pass system. The EU uses a three-pass slow system. Japan has both either one- or two-pass system, similar to the fast spectrum, and the Russian Federation also has the one- or two-pass system concept option.

So those are the key design parameters different between each concept.

So I mentioned about the majority of the function for the SCWR is for electric power generation, but because of the high temperature at the outlet, we can only use the steam from the outlet for cogeneration. For example, we can use it for hydrogen production and for oil extraction using through the steam-assisted gravity drainage process or get the heat out for the desalinization process or using the heat for process heat. So those are the options that we can introduce into the SCWR and that will help also further improve the efficiency of the concept of the reactor.

So from the concept development status, as I mentioned, Canada, the EU, and Japan have completed the development of the concept. And after we finish the concept, we usually invite international peers to review the concept and assess the viability. We have finished the development, but we still have some work to be done. And we have introduced the R&D program to actually help us to improve the confidence on the concept itself.

China and the Russian Federation are working on completing their concepts, China also plans to host a review of their concept with their international peers after the completion of their concept development.

And one of the key areas of interest is to perform a fuel irradiation tests. The reason why, as I mentioned previously, we haven't actually developed or established an in-core SWR facility for operation or testing, so preparing for a fuel irradiation test, that will give us the opportunity to acquire design and licensing experience for an in-reactor supercritical water cooled system. So the test itself will provide us with in-reactor test data for fuel, for cladding material, and also thermal efficiency, I mean thermal-hydraulics as well at supercritical pressure. The data that we obtain can also be used for code validation.

As I mentioned previously in the figure, the core size is actually quite big, the power generation from 2300 up to 4000 MWth. Except for the superfast reactor from Japan's concept, then it has lower thermal power generation, around 1600 MWth. But a lot of small remote communities actually require much less power than 1000 MW. So the flexibility of the SCWR concept allows us to adjust the core size to meet the needs of local deployment.

We are also interested on developing a small SCWR concept or even a very small SCWR concept, varying from 10 MWe up to 300 MWe to meet the local deployment needs. So this is something that we are starting to work on right now.

So from the GIF Technology Goals point of view, when it was set up in the beginning, they established several goals to achieve the advanced nuclear system: economics, for example, that will try to reduce the capital and operating costs for an advanced nuclear system; to improve the safety and reliability, which will allow us to strengthen the public acceptance of nuclear energy; helping in sustainability, which means that we can reduce waste and improve resource utilization; and also to enhance the proliferation resistance and physical protection, which will help us strength public confidence on nuclear energy. So all these technology goals we try to achieve in developing the concept. But on top of that, what we have to think about is how the concept has to be competitive with other power generation sources and systems in the local deployment area.

So in the following several slides, I will show you how the SCWR aligns with the technological development by GIF.

So economics, the GIF economic goal for the Gen IV system is having a life cycle cost advantage over other energy systems. That means it's lower levelized unit cost of energy on average over their lifetime. And also have a level of financial risk compatible to other energy projects and on similar total capital investment and capital at risk.

So when we look at the costs, below, if we look at the capital costs, the fuel costs, operation and maintenance costs in establishing the comparison for economics of different systems, and to evaluate the cost we look at the total capital investment cost, GCIC. That is actually the overnight capital costs plus interest during the construction. So if you have five years' construction, that means you have to capture the investment plus the interest during that construction phase.

And then we look at also what we call the levelized unit electricity or energy cost, LUEC. Generally, when we look at the cost evaluation, because this is a new system, it's an advanced system, there's a high degree of uncertainty surrounding economic estimates for this kind of concept. So in many cases, we base it on the current knowledge on the PWR or PWR system, and then apply some quantification to assemble the costs for the estimate.

Because the SWR has high efficiency, that means that we need fewer plants for the same demand of power or energy, so that in fact already reduces the capital or operating costs in general.

So the pressure vessel type SCWR economics, actually the EU SCWR concept, what we call the high performance light water reactor, has assessed the capital costs against the advanced boiling water reactor, and in here, they follow the GIF economic assessment guidelines, and that's before they actually proposed the model for use in the economic assessment.

So here's the figure that's comparing the total overnight cost for the advanced boiling water reactor plant and the high performance light water reactor. They are separated into different components in this area, and overall what they find is that total overnight costs for the HPLWR is about 20% lower than the ABWR.

They also look at the sensitivity of the capital and fuel costs on the electricity generation cost, and what they find is the capital cost variation basically affects the electricity generation cost but only in the short term, and then over the long run it diminishes, and then the effect becomes less significant.

The fuel cost variation also affects the electricity generation cost over the depreciation period, and also the difference is reduced with time as well.

So the pressure tube type SCWR they also compared with the advanced boiling water reactor, and the information that they collected is from the Tennessee Valley Authority Bellefonte Site. And the cost that they provided for that site was actually presented in 2005, so after 2011, the Fukushima incident, a lot of safety systems have been introduced into the ABWR as well, so those costs haven't been included in the calculation.

So they apply actually the chief economic modeling too, and including the uncertainty into the calculation, and what they find, they compare the TCIC, or the total capital investment cost, and they find that it is compatible with the advanced boiling water reactor.

And the levelized unit electricity cost actually is higher for the Canadian SCWR compared to the ABWR. It's mainly because the fuel cost is higher. As I mentioned before, the calculation, the uncertainty, is also very high.

So this is a table where they compared the Canadian SCWR concept with the other systems, including ABWR and AP 1000 and Summer AP 1000 and Vogtle.

You can see with the ABWR, it's compatible, 44 to 41, and then compared to AP 1000, that is slightly lower. The Vogtle AP 1000 is quoted in the reference, it's a bit low, and I think they have revised here the cost right now in the system.

So for LUEC as a comparison, again, like I said, SCWR is slightly higher than the ABWR, but it is also compatible with the other quoted price or LUEC.

But the economics right now, at this point it's still at the concept level and it could be also improved, for example, if the ASWR is used, because it's using the plutonium thorium fuel, if you consider the burnup for the excess plutonium, then that will improve the economics, or because of the high-temperature outlet, it can also include the cogeneration and that will also improve the economics as well. So those are the options that we can introduce.

From safety and reliability and the GIF Safety and Reliability Goals for the Gen IV Systems, it excels in safety and reliability having a very low likelihood and degree of reactor core damage or eliminates the need for offsite emergency response.

And the Working Group actually developed what they call the Integrated Safety Assessment Methodology, and they introduced several tools to assess the safety characteristics. The tool itself, there are five different tools, and you need to apply based on the maturity of the design of the concept. So the qualitative safety feature review or the QSR, and Phenomena Identification and Ranking Tables, PIRT, Objective Provision Tree, OPT, and then the Probabilistic Safety Assessment, the PSA, and Deterministic and Phenomenological Analysis, known as DPA. So it depends on the maturity of the design, you can apply different tools in the assessment. So in the concept phase, usually you can apply the QSR, PIRT, PSA, and the deterministic analysis.

So the safety and reliability of the characteristics of the SCWR actually is quite similar to the current fleet of reactors, for example the pressure vessel type SCWR is compatible to the pressured or boiling water reactor. The pressure tube type is compatible with the pressurized heavy-water-type of reactor.

And the safety requirements, there's a difference between the SCWR and the current fleet of reactors. The current fleet of reactors focus on maintaining the coolant inventory in the system, but in the SCWR we are focusing on maintaining the coolant flow rate through the core because of the direct cycle, as you know, with recirculation in the core, so the feed water flow is very important for the SCWR.

So for the pressure-vessel type, Japan's super light water reactor, they actually performed the safety analysis using the deterministic analysis. They covered the key postulated accident scenarios, and what it find is a 15% break of the loss of coolant accident is the limiting event for them. The predicted peak classing temperature in the measure that is around 1000°, that is about 350° above the steady-state value.

The also performed what is called the Simplified Probabilistic Safety Analysis. They quantified the core damage frequency at around 5.1E-7 for the large-break loss-of-coolant accident.

The European Union also performed the safety analysis for the high performance light water reactor. They applied the deterministic analysis to a couple of selected postulated accident scenarios only. What they find is a total loss of feedwater is a limiting event for the HPLWR. The peak cladding temperature to be taken at around 910°C. The (order) has introduced a number of passive safety systems to enhance the safety characteristics of the high performance light water reactors.

From the pressure tube type SCWR, they actually follow the GIF methodology, and they performed a Qualitative Safety Features Review, and covering five levels of defense-in-depth provisions assessed in the case. And they also performed a PIRT, a Phenomena Identification and Ranking Table, and in here, this is the table that they can take a look at the rank of importance, high, medium, and low, and instrument, and the status of the knowledge from 1 to 4; 4 is a lot of knowledge and then 1 is a lack of knowledge.

So they identified 30 knowledge gaps in this establishment, and the majority of the knowledge gaps are related to new material use in the core, especially the ceramic insulator that's insulating the high-temperature coolant from the low temperature pressure tube and also the moderator.

And they applied a deterministic analysis covering a lot of postulated accidents scenarios, and they find that one, the coupled loss of coolant with loss of emergency core cooling accident is the most limiting cladding-temperature event for this concept. The peak cladding temperature is predicted at around 1175 is close to the design limit, but it's still below the melting temperature of the cladding.

They also performed a Simplified Probabilistic Safety Analysis and basically looked at the core damage frequency for several postulated accident scenarios, small break LOCA, large break LOCA, and loss of class IV power, and they find that the probability of core damage is at least one order of magnitude lower than other reactor systems. But like I said, this is a simplified system, and then this still has a lot of uncertainty and needs to quantify.

So sustainability, the GIF Sustainability Goals is to generate energy, sustainability, and promote long-term availability of nuclear fuel, and also minimize nuclear waste and reduce the long-term stewardship burden.

So at this point, GIF has not developed or established a common methodology for assessment yet, and then we only base it on the knowledge that we have and then develop a sustainability matrix in the assessment. But in general, nuclear energy actually is one of the lowest sources of greenhouse gas emitters. It's 20 to 30 times less than fossil fuels, including natural gas, so in that case, actually, it generates energy with minimal damage to the environment.

And we applied the sustainability metrics. It meets the clean air objectives. I think usually this is a nuclear system resolving or meeting this kind of objective. And we promote long-term availability of systems for more effective fuel utilization, to minimize and manage nuclear waste, and reduce the long-term stewardship burden of nuclear waste, and also, improve protection for public health and the environment. And those are the metrics that we use to establish and fulfill the sustainability characteristics until GIF comes up with a common methodology.

So because the SCWR has high efficiency, like I mentioned before, it increases the power output for the same amount of fuel input, so in reality you reduce the fuel utilization. And then you also reduce the waste heat from turbines and condensers; that means that is good for the environment. And it promotes other effective fuel utilization, minimizes nuclear waste, long-term availability of systems, and environmental protection based on efficiency improvement.

And the Canadian SCWR implementing the advanced thorium fuel cycle is actually improving the sustainability because thorium is more abundant than uranium in the world. And the use of plutonium thorium fuel extends the natural uranium resources, so you don't need to add any uranium into the system anymore.

And use of thorium fuel produces a future usable fissile inventory of U-233, so you can continue using U-233 replacing the plutonium and then use it as the breeder to breed thorium. And also, lowers the long-term gamma of used nuclear fuel, and also reduces the amount and decay power of the high level fuel as well, based on the advanced thorium fuel cycle.

So proliferation resistance and physical protection, and you look at that, the GIF Proliferation Resistance and Physical Protection Goal is to be a very unattractive route for diversion or theft of weapon-usable materials and provide increased physical protection against acts of terrorism. So this is, the cycle itself is from the fresh fuel coming, to the fresh fuel storage, to go for the reactor, the spent fuel storage and go to the shipping and receiving of the spent fuel.

So the methodology developed by GIF on the Proliferation Resistance and Physical Protection actually took a look into the proliferation resistance measures and also the physical protection measures. So we use those two measures to establish the characteristics for the SCWR.

So we identified the main threats for these two. For the proliferation resistance threats, it's diversion of fresh and/or spent fuel, and also concealed production, that means the misuse of the fuels or the spent fuel. On the physical protection threat, it's a sabotage attempt to cause radiological release. Those are the two threats that we have identified.

You look at the SWR compared to either a BWR or PWR, the SW actually has a very small footprint, smaller than the current fleet of reactors. So the smaller footprint, actually we can enhance the opportunity for physical protection because there is less area to cover.

So most concepts that we have introduced are based on familiar technology from the safeguard point of view, for example, that we use a thermal spectrum using batch fueling, using solid fuel, using light water coolant, those are familiar technologies for safeguards. If you use thorium, the advanced thorium fuel cycle, it enhances also the characteristics on proliferation resistance and physical protection because thorium is fertile, it's not fissile, and so it has lower plutonium production from the thorium, and it's a production of U-233 and U-238 together and it's very difficult to separate the two, and the spent fuel contains also deep burn of plutonium and U-233 mixed with U-232. And the radioactivity of the spent fuel is a barrier to diversion of the spent fuel assembly as well because of this high level. So I think that those are the key areas.

So I covered the alignment with the GIF Technology Goals for the SCWR. Now I will talk about some of the design challenges we have in developing the concept. I mean, because it's a high-temperature high-pressure reactor, one of the challenges is to get the material. For the in-core and out-of-core components, what we find is no single alloy that has sufficient information to confirm the performance for use in the SCWR component. But we can adopt a lot of information on the experience from the current fleet of reactors and the fossil fuel power plants. And we have to establish different acceptance criteria or requirements for the corrosion on the nuclear power plants. And because of the high-temperature operating conditions in the core, we need probably a thermal or corrosion resistance barrier in components that have a very high-temperature grading.

So one of the key areas that we find is challenging is the cladding. Because it's high-temperature operating conditions, the zirconium-based alloys that are currently used in the reactor are considered not viable materials, so the majority of the fuel concept actually adopts the stainless steel or nickel-based alloys as their candidates.

So to codify those materials for use in the SCWR, we have to demonstrate the performance in the key areas, for example, in corrosion, stress corrosion, cracking, strength, embrittlement or creep resistance, and the change also in the dimension and microstructure during the operation duration.

But keep in mind that in many cases the cladding still in the core is over a very short time, and every year we'll take it out and move the fuel into a different location, and then we have to, after three batches we'll take it out from the core. And on top of that, we need to quantify the irradiation effect on this kind of key technology area.

So in this figure I'll show you the example, that is measuring the weight gain for the stainless steel 316 and 316L at the supercritical SCWR condition. This a function of the weight gain, it's a function of temperature in the autoclave, so it varies from around 250 up to 550°. You can see the weight gain. This is simulating the corrosion characteristic. It's actually increasing quite fast when you go to a high temperature.

So I mean, we can test it up to higher than 550 but we still have the challenge and the testing at a higher pressure and a higher temperature, for example, we don't have any autoclave that's beyond 700° that can test the corrosion and those things. We started constructing the new loop that can withstand 850, so hopefully we can get some data at the higher temperature.

And we also need the reactor or in-reactor test for irradiation defect, and we are looking at the other effects, other irradiation effect, using the accelerator as well, so hopefully we'll get some information to quantify the impact, the irradiation on those key areas.

You look at the SCWR material selection for example, this is the figure that shows you the complete core configuration, the core, the turbine, the lower pressure, low-temperature turbine and the feed train area. So you look at, first, the material selection in the low temperature area and the feed train can be based on the PWR or the fossil-fired power plant, but we mainly did some optimization to minimize the impurity transfer from the core through the turbine to this area.

So the focus is actually that in the high core, in the high-temperature turbine area because that's the area outside the bound of the current OPEX, so we need to focus on those areas and develop and establish a material candidate.

So we used a different methodology to establish the rank of the different material for cladding material, and we looked at five different materials in this area. Alloy 800H, stainless steel 310S, 625, 347, and 210, and looked at the properties, for example, corrosion, oxide thickness, creep, ductility, and strength.

So we established the different properties, for example corrosion, we will see that we have available data that this alloy will meet the corrosion criteria. The orange area in this area, that means that we have some data available that may or may not meet the performance, and the red area, that means that we have data to show that the alloy will not meet the performance criteria. The gray area is the area that we find that we don't have sufficient data to quantify the effect on this property or this cladding material.

So if you look at it overall, the alloy 800H, for the purity, none of the alloys seems to have the sufficient data to cover the ductility property, so you can see the alloy 800H and 625 seem to be the best with sufficient information to justify them as a good cladding material candidate.

So I mentioned about the corrosion test at high temperature as one of the issues, and we are looking also to find out whether we can use low-pressure superheated steam as a surrogate condition to test at high temperature or high pressure. So they performed some corrosion tests on the austenitic stainless steel, 310, 304, and nickel, iron based A286. They performed the test at .1 MPa, 8 MPa, and 29 MPa, at 625°C for 1,000 hours. What they found is they found a single-layer oxide formed at low pressure, and a dual-layer oxide formed at high pressure, 8 MPa and 29 MPa. And this is followed by a chrome-depleted area into the austenite substrate.

But if you examine closely, actually the composition for the inner oxides at 8 MPa and 29 MPa are also chrome rich, and thus the layer is actually similar to the 0.1, the low pressure exposure.

So based on the test results, they find that using superheated steam may provide a qualitative information similar to testing at the 25 MPa condition.

So for chemistry, the change in the chemistry, the chemical property is at first drastic and expressed at a supercritical point. So the SCWR at the in-core radiolysis characteristic is quite different from the current water cooled reactor. This is a figure that shows the density variation with the temperature, so you can see that at low pressure the density varied, you have a sharp drop at the saturation point. So in that case, the majority of the reactor actually at the density, operates at the low-density region, I mean the high-density region, so it doesn't, except for the accident condition, then you drop to the low-density region in the PWR.

But the PWR you operate, you actually pass through the saturation point. So from the supercritical SCWR at the 30 MPa for example, you see the variation that still has a rapid drop at that (tendency) at the supercritical point but more or less it is continued because you don't have interface change.

But for the mass deposit and the (flow) to this area, you can see it's very low at the low temperature, and then when you reach the supercritical point, the deposit mass is drastically increased, and then when you go to high temperature, it is decreased again.

So those are the characteristics of the material behavior affected by the current chemistry of the system. So we need to identify an appropriate water chemistry to minimize the corrosion rate, stress corrosion cracking, and also the deposit into the fuel cladding or the turbine blade. The need for us is to establish a chemistry control strategy.

I will show you the same figure that we showed before in the material, but look at the chemistry control area. So you can see, in the system it's only this part is supercritical. The rest of them is just a low-temperature and low-pressure region. So in the chemistry here, in the feed train area, it is determined what is the concentration of impurity that enters into the core, and thus there is a focus on the feed train.

So what we propose, you can use the full flow condensate polisher like BWRs to remove any impurities from the fuel cladding, and you treat using oxygenated treatment for the feed train in this area, and then possibly introduce hydrogen addition upstream of the core. This is to control the oxidizing species in-core so that you can minimize the activity transport to the turbine or through the cladding.

So for thermal-hydraulics, one of the key differences between the SCWR and the current fleet of reactors is the lack of feed train because at the supercritical pressure, the coolant actually acts as single phase. There's no boiling, so no phase change. So whatever the cladding, the use of the tradition CHF criteria, or critical influx criteria, are no longer valid or appropriate for the SCWR, so the limit that we use is always based on the cladding temperature or the fuel temperature as the criteria.

So we need to predict the heat transfer accurately because of the sharp temperature or the property variation along the supercritical point, so you can see, this is the figure that shows you the density variation, thermal conductivity, specific heat, of viscosity, and as a function of bulk flow temperature. So this is a pseudocritical point in this area. You can see the property drops sharply in this area but not as sharp as this ??? when you have phase change. This is a variation of the density and the viscosity. The specific heat on the other hand that you see has a low value, has a low temperature, and then jumps up and then increases sharply at the pseudo-critical point, and then after that it decreases and then goes to the low value again at high temperature.

So because of this kind of sharp variation of properties, you have a serious significant impact to the heat transfer characteristic.

So we have collected a large amount of experimental data to support the heat transfer calculation. We have a lot of tube data. The majority of tube data actually comes from the fossil power plant experiment. So those are data that, because the (fire) tube from the fossil power plant is usually bigger, so they may not be directly applicable, but at least they provide some information to understand the phenomena. And we also collect data for the 3-, 4-, and 7-rod bundle subassemblies.

And at this point, because we are still in a conceptual phase, testing of a full-scale bundle is considered premature because we are still optimizing the fuel assembly or improving the fuel assembly heat transfer characteristic. But on top of it, most of the experiment is just operating or testing with uniform heating or reference cases. We don't have any data for the separate effects, for example, non-uniform power distribution, axially or radially, and those are the information we need to do the assessment of the heat transfer characteristics, to establish the power generation, and the safety criteria.

This is a sample of the database that we have collected. We have annuli data, tube data, 4-rod bundle, 3-rod bundle, 7-rod bundle in water, CO_2, carbon dioxide, refrigerant, and in different configurations. So I'm not going to go into detail on the slide but just show you a few figures on the heat transfer experiment that we have done.

We have done the 4-rod bundle in high pressure water flow, so the flow condition is 25 MPa, 1000 kg/m2s flow and the temperature, 416°C, and the rod itself they have equipped with a movable thermocouple in the one rod and a fixed thermocouple in the other rod, and these are the data without any spacer effect.

And this is a comparison of the circumferential heat transfer, the low temperature that they have. This is a movable thermocouple in a very small angle increment, and those are the fixed thermocouples. You can see that both agree quite well, very well, between the movable and the fixed thermocouple. And that shows that the rod performance is very good.

And in this figure, they compared the low temperature measure between the rod with no spacer and the rod with a spacer. In general what we find is that the rod with a spacer usually has a lower temperature, in this area, as an open symbol, and a lower temperature, except when the spacer is located in that area because the test, the rod itself is directly heated, so in that case, the spacer also introduces heat to the system, and that's the reason why you have a localized peak in the temperature. But in general, what we find is the spacer is effective in handling heat transfer.

In this case they measure the 3-rod bundle in a carbon dioxide flow, and this is the axial low temperature variation with one of the rods at the different axial location. And they also compared the difference between the grid spacer and the wire-wrapped spacer. Again, the wire-wrapped spacer seems to provide a systematic improvement in heat transfer or lower to (bore) temperature, but at the grid spacer they introduce the improvement in heat transfer at the grid area, but the improvement actually decreases very fast downstream of the grid spacer.

For safety, the challenge is we have to demonstrate safety design effectiveness in the system. We at this point have collected some of the transient experimental data, but still we need more data on this area and the critical part, especially for the pressure transient through the pseudo-critical point.

We also have collected a large amount of data at the supercritical water test for critical flow that will help us to design the safety/relief valve and also the depressurization system and also support the large break LOCA analysis that will be used in the safety analysis.

And because the system has a very strong couple link between the neutronic and thermal-hydraulic and it's also susceptible to dynamic oscillation, we have collected a large amount of data to look at the stability of the flow in the system.

And one of the challenges that we have is to quantify the applicability of the safety analysis tool, and we need data to validate the tools because the majority of the tools have been validates for subcritical conditions only, so we need to have integral test data at the super supercritical condition to validate this code.

So some of the transient experiments that we have collected so far, for example in the 4-rod bundle, and you look at the pressure transient, a drop from a supercritical pressure down to a subcritical pressure, what we find is the temperature, the cladding temperature or the wall temperature at the rod is actually quite low at the high pressure condition when you achieve or approach the supercritical point, the temperature actually increases drastically, and this is similar to the (thin) boiling conditions at the subcritical region.

And we also look at the 3-rod bundle. If you look at, this is the power transient simulation, it's basically, you drop the power by 15 for some time and then increase it back to the normal condition, you can see that once you drop the power, the system parameters also varies with the changes, and then it takes some time for the flow for example to stabilize, and the temperature, actually the wall temperature follows quite closely with the changes.

And the other test that we have is the full transient. It's dropping the flow by 15% for some time, and then bring it back up, and you can see that the power is not affected by the transient. It's because it's stable, but the pressure (oxalated) when you drop the flow before it becomes stabilized. The wall temperature for example in the (two cases) corresponds very closely with the flow variation. So that is the information that we have so far.

On the physics point of view, actually the neutronic design has an impact on safety, economics, and all the sustainability and proliferation resistance and security. The coolant changes actually from liquid-like to gas-like fluid over the core, and then it's affecting the absorption or moderation of the core characteristic.

What we seem to find is the characteristics, actually the physics are similar to those of the current fleet reactor, except for the differences in geometry, temperatures, and properties, and you look at the very strong coolant, the moderation effect, then we definitely need to embark on the 3D neutronic, the thermal-hydraulic coupling.

One of the challenges we are facing is to establish the accuracy of the physics codes for the (harder) neutron spectrum and a higher fuel temperature and a moderator temperature And we also need the data to validate the physics codes, and those are the challenges we have.

As I mentioned, the research and development is based on very strong collaboration between the different groups of partners, and the collaboration actually allows us to leverage resources and expertise from different partners to expedite the development. We have a strong collaboration between the Generation IV International Forum in the Systems Steering Committee and also the Project Management Board. And we also have strong support from the International Atomic Energy Agency, who organizes the coordinated research program to involve non-GIF member states into the development.

And we also established a bilateral agreement between different groups, and before, for example, China signed the project arrangement, we have a bilateral agreement between the EU and China, and Canada and China, to work together on the common interests.

And we have a lot of exchanges of information in the international symposium on SWCR. We just completed the eighth symposium now in Chengdu, China, and we already have strong participation from GIF and non-GIF members in the symposium.

Under GIF, we also organize the information exchange meeting between different partners to help us to disseminate any information between partners. And as I mentioned, the IAEA has the Coordinated Research Projects and Technical Meetings, and that's also the venue for us to interact with non-GIF partners in the supercritical water heat transfer for material development.

So in summary, we have several SCWR design concepts, both the pressure tube type and the pressure vessel type. We have a direct thermal cycle that can simplify the concept design and also reduce cost. The thermal power generated from 1600 MW for the fast reactor in the Japan concept to around 4000 MW for the thermal concept. And the thermal efficiency is higher than 43% and as high as 48%.

We have a thermal spectrum, fast spectrum, and mixed spectrum concept. And we use light water and heavy water moderator for the thermal spectrum, and a solid moderator from the blanket fuel for the fast spectrum. And we've seen that there is some similarity between the thermal spectrum core concept now emerged. For example, the Canadian pressure tube reactor is using the inner (plenum) similar to the (first) vessel but not in the active zone.

And then the subdivision of the core into a different zone that is similar to a pressure channel or a pressure tube. So those are the cases that we have emerging.

We still have design challenges and we identify them. Some have been resolved and others are still being addresses.

I think this concludes my presentation so I provide some references at the back of the presentation. Please feel free to look at them. And thank you for your attention.

Oates: Thank you, Laurence. If you have questions for today's presenter, go ahead, please, now, and type them in the Q&A chat pod, and while those questions are coming in, we'll give you a sneak peek at the upcoming presentation.

On the 27th April, a presentation by Prof. Peterson with UC Berkeley, Fluoride Salt Cooled High Temperature Reactors; 23rd May, a Molten Salt Reactor presentation by Dr. Elsa Merle with CEA in France; and in June on the 20th, Lead Fast Reactor, a presentation by Prof. Craig Smith with the US Naval Graduate School.

So I do see some questions, Laurence. Do you see...?

Leung: Yes, I see it. Okay, and the first question is, you mentioned that the SCWR can be applied also for cogeneration – for hydrogen production. Which way can be produced hydrogen when...? I cannot see the whole thing. That's a problem.

Oates: On the right-hand side of your chat box, there should be a scroll bar that will help you move down.

Leung: Yes, okay. Anyway, I'll go to 17. So sorry. In the hydrogen production, because of the high temperature coming out and from the turbine, we can use the copper-chlorine cycle (for hydrogen production. Is that okay? This is by adopting the higher copper-chlorine cycle which requires only the low temperature as compared the ??? cycle. So that's the cycle that we are planning to use. Okay? Next one.

Slide 26 states that a simplified PSA indicates that pressure tube type has a core damage probability that is at least an order of magnitude lower than...

Sorry, I still just can't see the question.

Oates: It says, are you reading the one from Craig Smith?

Leung: Yes.

Oates: Slide 26 states that a simplified PSA indicates that the PT type SCWR has a core damage probability that is at least an order of magnitude lower than the other reactor systems. What are the other reactor systems that this refers to?

Leung: Oh, we refer to the current state of reactor. Okay? This is the current state of reactor comparison.

Okay, we've got, how would pressure tube SCWR fuel cost be affected if UO2 were used instead of Pu-Thorium?

Good question. We did a comparison between the UO2 and Pu-Thorium as well, and the UO2 is slightly lower, but it does not drastically change, but the uncertainty is smaller so that is different. A good question.

And the next one, what are the advantages for Canada SCWR to use heavy water?...

So thinking about. Okay...

Being able to use natural uranium fuel or non-enriched uranium fuel?

Yeah, we cannot use natural uranium fuel in this case. And the advantage actually is the neutron economy. Bringing the heavy water, actually that helps us on the neutron economy. And we can also use light water, but we find that heavy water is better so that we can use lead enrichment and then we can use also thorium as well.

The next one, interesting presentation, thank you!

Oh, thank you, thank you, Mark.

On the scheduling note, please clarify that the ??? is scheduled for June 12, not June 20. Okay. [*laughs*] That's for Berta.

Oates: Yes, I apologize. Thanks, Craig.

Leung: Okay, thank you, Michael.

On slide 12, can you explain more the heat cycle on Canadian SCWR?

Slide 12. Okay. Yes, this is slide 12, can you explain more the heat cycle on Canadian SCWR?

The Canadian SCWR is here, so let me pull up this thing. So what you have in here, the fuel, this is the coolant coming in and through the nozzle in here acting as an orifice, you adjust the orifice to match the flow that is coming (to match in) power, so the flow that's coming in the outside is getting to the center, and through again another set of nozzles, so through this, it comes down, and then through the center area, in this area, and then reverses direction, and then coming up through the gap, this is the fuel rod, in this area, and then passing through this area, and then this is the steam coming up and then going to the outlet header.

So this is the case that the pressure tube is here and the fuel itself, you have an insulator, and then this is a fuel rod, so in that inlet flow coming down, the blue area is the tube, and then reverse at the bottom, and then coming up through the fuel rod.

I hope that clarified the heat cycle on the thing.

And on slide 21, what does the purple circle represent, only denote 8000 ppb?

Okay, slide 21. Is that the figure, slide 21? Ppb, 8000 ppb. Sorry, I don't find this 8000 ppb in here. Guopin, on slide 21, the purple circle, I don't understand the 8000 ppb. Sorry.

Why thorium is suitable for SCWR?

Another question, why thorium is suitable for SCWR. Thorium is suitable for a pressurized heavy water reactor and also suitable for SCWR. I don't understand the question.

Thorium is the kind of fuel. It's fertile, so you need to have either the UO2, enriched UO2, or reactor-grade plutonium to start the process. And I guess if you have enough then the neutron will start the process and it's suitable, and we did the calculation already. It's applicable for the SCWR.

Thank you.

Oh, slide 31.

Oates: That's the one with the circle. Right.

Leung: Oh, okay. Oh, the circle is different, oxygen I think, ddp on the oxygen, therefore in the autoclave, that's the test. It's a different condition that they've introduced, so that's parts per billion.

Okay, yes, I hope that answered the question. It's a different condition that we introduced into the autoclave.

Thank you. I don't see any more questions. So...

Oates: That looks like that is the end of the questions at least so far. I will go ahead and make note of the correction again in June, the presentation on Lead Fast Reactors is scheduled for the 12^{th} June, not the 20^{th}. I apologize for that typo.

Leung: Thank you, Muhammad, and I really appreciate you got to participate in this webinar. I hope that I provided some information for you to move on with your research, and if you have any questions, feel free to drop a line to me. And I hope that we can discuss and continue the dialogue.

And as I mentioned, the SCWR actually is worked through collaboration. I hope that we get more partners and more people working on it so that you can help us to expedite the design.

I got another question. You said that the reactor plant for SCWR is also finished, so have you developed neutronic calculation after the shutdown of reactor, and how long for the shutdown period?

Well, we developed the plant concept. I think that we still have some lingering work to be done. And I'll mention, this is the concept phase, and we'll continue working on the startup/shutdown calculation as well. So it's still continuing. Thank you, Najoua.

Okay.

Oates: Thank you again, Dr. Leung. It's been a pleasure working with you through this webinar presentation and getting ready for it. I appreciate the effort to put together the slides. Thanks to everyone else who has attended. Your attendance makes these very worthwhile and your interaction during the Q&A is always appreciated.

Amanda, thank for your help on the back end. And Patricia, thank you for your leadership through the Education and Training Task Force in putting these presenters, getting the people gathered to do these presentations of course is always very appreciated. I guess with that we will conclude and wish everyone a good day.

Leung: Thank you.