

Materials Challenges for Generation IV Reactors

Dr. Stu Maloy, LANL, USA

Berta Oates

Welcome, everyone to the next Gen IV International Forum webinar presentation. Today's presentation is on Materials Challenges for Generation IV Reactors. Doing today's introduction is Dr. Patricia Paviet. Patricia is the Director of the Office of Materials and Chemical Technologies within the DOE Office of...

Sorry, Patricia, your signature is a little wrong. I butchered that a little bit.

Patricia Paviet

It's fine. Nuclear Energy.

Berta Oates

Yeah, Nuclear Energy.

Patricia Paviet

Okay. Thank you so much, Berta for the introduction. It's my pleasure today to have Dr. Stu Maloy from the Los Alamos National Laboratory. He is a team leader for MST-8, the Materials at Radiation and Dynamic Extremes. He has worked at Los Alamos for 28 years and he is the Advanced Reactor Core Materials Technical Leader for the Nuclear Technology Research and Development's Advanced Fuels Campaign and the NEET Reactor Materials Technical Lead for the DOE, Office of Nuclear Energy.

He earned his Ph.D. in 1994 in Materials Science from Case Western Reserve University and he is a registered PE in Metallurgy. He has applied his expertise to characterizing and testing different materials and ceramics in extreme environments. He has also more than 190 peer reviewed technical publications and numerous presentations. And without any delay, I am going to give the floor to Stu. Thank you again, Stu for volunteering to give this GIF webinar.

Stuart Maloy

Well, thank you, Patricia for that very nice introduction. It's a pleasure to be here and welcome, everybody, good morning, afternoon or evening from wherever you are looking in from. Let me just start off with a brief outline of what I'll talk about today. I'll give a little background on radiation effects in materials and then talk about radiation effects in specific materials for nuclear reactors, specifically Face Centered Cubic and Body Centered Cubic alloys. And then I'll get into the specific Gen IV reactors and their materials and challenges for the different operating

conditions of these reactors. In the end, I'll summarize that and then, summarize some of the performance issues.

So, I think most people here know that materials in nuclear systems can fail, especially in some of the harsh conditions that they must see. For example, in the upper left hand corner the fast reactor ducts that saw very high exposure. You can see up to 52 to 34 dpa exposure actually failed when removing them from the reactor. That's a high dose reactor issue. And then other issues from corrosion as you can see, the Davis-Besse reactor vessel had degradation or grid-to-rod fretting typical Zircaloy clad fuel cladding and crud as well can be an issue as well. So, clearly, there are some materials issues you can run into in reactor systems.

What I want to do is just summarize the basics of radiation damage. As most people know, when a high energy particle comes into a material, into a metal lattice, when it interacts with that, it can form a large displacement spike as you can see shown here. That's just picoseconds after the actual interaction with the material. But then afterwards when it relaxes, you end up with a large number of interstitials you can see around the displacement spike, and they can see is surround [ph] the center. And that's what leads to damage within the material that can cause changes in mechanical properties with high dose radiation.

So, you can separate this into two things that happen during radiation damage. You can have transmutation occur, so you can actually activate the material, of course. And also you can produce new elements within the alloy. And many times such as for example in nickel transmutation, during that decay of the nickel-58 atom, you can give off alpha or helium into the material which can then lead to other deleterious effects in the material.

And then the second change, the materials property changes due to atomic displacement. As I pointed out just before, when a high energy particle comes in, it can lead to displacements which cause vacancies in the material and the atom goes into an interstitial and then causes self-interstitials in the material as well. And those vacancies can collect to form voids in the material which leads to swelling. Because of the high amounts of defects in the material, you can get increased diffusivity which leads to local segregation. And then you can even precipitate out new phases or cause amorphization of phases under radiations. All these can lead significant property changes in materials under radiation.

An example that most people can sort of think of with respect to a high energy cascade coming into the materials. Think about playing pool and when the initial cue ball hits your set of pool balls which would be like your atoms in the lattice, it causes displacements as the energy is

transferred from the initial high energy neutron or scattered neutron to the actual particles within the lattice. And then that creates a cascade of defects including vacancies interstitials within the material.

And what's important in the end really is the ultimate fate of the point defects. So, as shown here, you have the initial event of the particle interacting with atoms in the material and causing vacancies in self-interstitials. Now, depending on where the vacancies and interstitials are, you can get some recombination, and also depending on the temperature you can get recombination. That's really not an issue, but then you can also run into clustering which can lead to voids in the material or in some cases you can get absorption of the vacancies and interstitials at sinks. And that can lead to increased segregation which can affect mechanical properties in the end as well.

And as shown here down below, you can see right after that initial cascade, you have a very large number of defects in the material, but only a small time later it's the recombination of those defects and what's left that actually causes the changes in mechanical properties. So, that's really what we are worried about in looking at how a material can withstand high dose radiation in a reactor application.

And so, this slide just shows that when you sort of step up in scale, the actual changes in properties happen at the mesoscale. So you have different self-interstitials and different atoms perhaps made under radiation which then lead to dislocations within the material which can at a larger scale lead to interactions with the buildup of defects at grain boundaries or different places that can precipitate out under radiation. So, all of those are eventually what you are worried about with the changes in mechanical properties in the material.

So, what are typical alloy compositions? Just as a background, before I get into talking about the different material changes, austenitic stainless steels are used in many different applications within reactors. So 316L and 304L are high chrome high nickel alloys or Inconel-718 which is a high nickel, also chrome alloy. And Alloy-600, also a very high nickel alloy. So, these are all austenitic or face centered cubic alloys.

Ferritic steels, which also have significant applications to nuclear reactor components have a body centered cubic crystal structure. And typical alloys are those such as modified 9-chrome-1-moly, which is a high chrome but you can see there is no nickel in the alloy, so you stabilize. That way you don't stabilize austenitic face and it's a fully ferritic alloy. And HT-9 as well has higher chrome than the modified 9-chrome-1-moly, but also about 0.2% carbon. And that's how you get a ferritic martensitic microstructure within those materials. And then, zirconium alloys of

course are also typically used as cladding materials in thermal reactors because of its low thermo neutron cross section.

So, those are typical materials. To summarize radiation effects in metals, as I already sort of pointed out, typically what happens in metals is you get basically Frenkel Defects, self-interstitials and vacancies. And on the right are diagrams of body centered cubic unit cell and a face-centered cubic unit cell, and the close packed slip planes are these boxes and triangles that are shown here. They don't have any charge compositing defects. There is little effect from gamma radiation and amorphization is typically, uncommon unlike what you can see with a ceramic material. The close-packed structures are the face-centered cubic, the body-centered cubic and then, with the zirc alloy for example, there is a hexagonal close packed material.

A very nice paper by Zinkle and Ghoniem shows a way to understand radiation effects in different materials over a class of different alloys, BCC alloys and FCC alloys. And you can sort of look at them based on their melting temperature. And what I am going to center on in the next slides are a lower temperature region where vacancies are immobile and interstitials are mobile, resulting in interstitial clusters and loops and small vacancy clusters that are formed within the material and results in significant hardening and can also result in reduction in ductility.

At medium temperature, which I won't talk about much in my talk, but at medium temperature which is typically the operating temperature that you'd like to use these materials, you have increased vacancy mobility and it results in more self-annihilation of defect. So, you get more recombination of defects and so you get less hardening in that region. But at high doses, you can run into void [ph] swelling in those areas as well.

And at higher temperature, you can run into issues such as with increased mobility in the material, you can run into issues such as creep. Or if helium is formed, it can lead to helium embrittlement, because it's now able to diffuse to grain boundaries. So, that's one way to think of how mechanical properties are affected in these different classes of alloys.

So, I am going to center on three different alloy classes that I'll talk about, 316L and 304L stainless steel, face centered cubic alloy, alloy 718, also face centered cubic, and then some ferritic steels and their exposure to radiation. So like I said, I am centering at lower temperature at 50 to 200 degree C in 316L and 304L. So, typically what you see under radiation in these TEM micrographs shown here is a buildup of interstitial clusters which eventually form interstitial loops with dose. And you can see that from 0.7, a fairly low dose dpa, there is a small number of loops within the material. At 3.8, you can see a much higher density and at 9.8,

you can see that the loop size is actually starting to grow a little bit as well.

And to summarize that here, the calculations done on this radiated material show that the loop density starts at about 10 to 22 per meter cubed, goes up to about 5 times in the 22, and then it actually starts to go back down. And at that point, like I was saying before, the mean loop diameter is starting to increase. So between 0.7 and 3.8, you actually start to saturate the whole dislocation density within the material and you can see that number actually is pretty flat from 3.8 to 9.8 dpa. While the diameter is going up, the loop density is going down, keeping the overall total dislocation density about the same.

And how does that affect the mechanical properties? This can be shown here in tensile curves versus a radiation dose. So, initial unirradiated material has a very high ductility, up to almost 40% uniform elongation. And then, after a very low dose, just 0.09 dpa, you can already see hardening in the material. At 1.1, it is continuing to harden. At 2.9, it goes up even further. So now you've gone from about 300 mpa up to about 800 mpa increase in yield stress and uniform elongation is starting to reduce significantly.

And that is the point where you sort of saturated the micro-structure with the defects and from there it sort of gradually increases in yield with increasing dose. As shown, there's a change between 2.9 and 9.3 dpa. Now, that can be summarized. One of the reasons why you run into significant reduction in ductility is because the micro-structure has a large number of defects in it as shown in this image here. And one set of dislocation starts to go through that micro-structure, it makes a channel. So, it makes a dislocation channel in that micro-structure. And that leads to a large load drop in the stress-strain curve. So, as you try to deform this micro-structure, you get significant localized deformation which reduces the uniform elongation as shown in these curves here.

And at the same point, you can also summarize the change in mechanical properties with dose in this graph right here as well. Yield stress as pointed out increases significantly after sort of very low dose radiation. I mean it actually started down here at around 300. It goes up very fast initially and then it sort of plateaus and increases with dose all the way out to, in this case tested out to about 10 dpa.

And along with that, reading this on the right hand side, uniform elongation drops significantly immediately. So, it goes from 50% to 60% down to about 20% and then it continues to decrease with increasing dose. To a point around 4 to 5 dpa, the uniform elongation looks like it's zero but it's about 1% here, all the way out to even high dose. And interestingly enough, even though the uniform elongation looks very low,

this material is still failing in a ductile manner. So, this SEM image shows a typical ductile fracture on the surface but because it's so localized, the deformation is so localized, you end up with very low uniform elongation in the material.

Now, Inconel-718 is also a face centered cubic material. It's response to radiation is a little bit different. This material, as shown here, has a large number of precipitates within the material which give it a very high yield strength. So, it has gamma prime and gamma double prime precipitates within the material. They are very fine distribution of precipitates in the range of about 10 nanometer in size, and they result in a very high yield strength material. But after only 0.6 dpa, all of these gamma prime and gamma double prime precipitates when irradiated at low temperature go into solution. And with increasing dose, they stay disordered all the way out to 13.1 dpa.

If you look at the plot of how that affects the yield stress or the stress-strain curve, it's a bit different than what we saw in the austenitic 316L stainless steel. In this case, you starts off with a very high strength material. Its yield stress is a little over 1000 mpa, has fairly good uniform elongation up to about 16%-17%. But after only half of a dpa when the precipitates become disordered, the uniform elongation decreases significantly. You get a little bit of an increase in yield stress because you are putting a large number of defects, interstitials and vacancies within the material. But with increasing doses, you continue to cause the precipitate to go into solution, the yield stress actually goes down. So, from 0.5 to 1.2 to 4.2 dpa, the yield stress is actually going down in the material.

And you can actually see that here on a plot of yield stress on the left hand side. That's the black dot. It increases initially but then with increasing dose it actually decreases out to about 11 dpa. And at the same point there is an abrupt reduction in uniform elongation, which is shown here, to a low amount. But once again, it's still failing in a ductile manner at least right now. And the uniform elongation is around 1% in that material.

Now, how does the fracture surface change with the radiation? In this case,, we've even tested. This was actually a component that was tested here out to 20 dpa, that's shown here, a 19.8 dpa. But in these stressed or these fracture surfaces, you see fairly ductile fracture at zero dpa, you see ductile fracture at 4.6. But at 10.5 dpa, you see some areas, as shown here, are flat or a brittle type fracture but most of the fracture is still a ductile fracture at that material.

But then at 19.8 dpa, there is an abrupt change. And in this case, I didn't show the stress-strain curve, but this material is actually breaking in the elastic regime and showing intergranular fracture.

So in summary, going back to the diagram that I showed before, how do we find materials that can actually reach very high doses, for example above 200 dpa. If you look at these classes of alloys, the refractory alloys are difficult to go to very high doses because they run into embrittlement issues due to interstitial pickup or swelling or specifically radiation hardening. FCC alloys look good but they run into issues because of void swelling after about 50 to 60 dpa as well as segregation. Zircaloy has issues with hydrogen embrittlement. So really what that leaves are the ODS, ferritic, and ferritic martensitic steels which I'll talk about here in the next couple of slides.

So here I want to summarize radiation defects in modified 9-chrome-1-moly and I'll talk about some outside [ph] dispersion strength in ferritic alloys as well in the end. So, modified 9-chrome-1-moly at low temperatures looks in a sense somewhat similar to the changes that we saw in, for example, 316L stainless steel, although it starts off with a much lower uniform elongation. In this case, it's only about 5% to 6 % uniform elongation. Now, these stress-strain curves are a little bit confusing to look at because I've got two different test temperatures at 160 and 50 degrees C which causes a slight difference in the stress-strain response.

But in general, at low dose 0.05 dpa, you still see about 4.5% uniform elongation, but at the 0.9 dpa now it's dropped fairly precipitously to around 1% uniform elongation and it stays like that.

At 2.9, this is a different irradiation temperature so it looks a little bit lower. But if it is the same irradiation, it would be going up. And yield stress, and then it continues to go up in yield stress all the way out to 8.8 dpa with maintaining a fairly low uniform elongation.

And that's also summarized here again. Yield stress shown on the left hand side in the filled squares increases significantly after fairly low dose and then it is a fairly gradual increase with increasing dose out to 10 dpa and at the same time uniform elongation is dropping off fairly quickly out to fairly high dose, out to 10 dpa.

What goes along with that is in body-centered cubic alloys, you have a ductile to brittle or DBTT, a ductile to brittle transition temperature. As you start to harden that material with low temperature radiation, it causes the ductile to brittle transition temperature to increase. In this case, it's going from about minus 150 out to at 9.3 dpa almost to room

temperature. So that can be an issue with low temperature irradiation of these materials.

And some work that we did here led out of Los Alamos was analysis of an ACO₃ duct that was made out of HT9. So in this case we were able to look at a component. What we did was to take advantage of the fact that along the duct the irradiation temperature goes from, right here on the right hand side, it goes up from about 380 degree C out to about 500 degree C as you go along the length of the duct. And the area where the fuel is the highest dose. And so the dose at that point right on the left hand side goes from about 20 dpa out to a maximum of about 155 dpa. So this was in the reactor for almost 7 years. And then as you move along the length of the duct and you move away from the fuel, the dose goes back down to even close to zero dpa at the upper part of that duct.

So we took advantage of that change in those different conditions along the duct and then, cut it along different locations listed here as 1, 2, 3, 4, 5 and then used those locations to test the change in mechanical properties with irradiation temperature and dose. So, shown here, you can see the changes in the stress-strain curves with irradiation temperature and in the upper left hand part are the lower temperature irradiated areas. I don't know if you can see this too well, but it's about 410 C and then about 380 and 370 degree C. So, as you are decreasing in irradiation temperature, you get an increase in yield stress and that's actually shown here as flat yield stress versus irradiation temperature. At the lowest irradiation temperature you saw the largest increase in yield stress over a control material.

And at the same time as shown in the other alloys, at the lower temperature that increase in yield stress results in a decrease in the fracture toughness as shown in these green dots here and it also results in, as you plot irradiation temperature versus transition, a ductile to brittle transition temperature, the DBTT is going up. And with decreased temperature you get an increase in the ductile to brittle transition temperature. But I should point out that in this temperature range here at 450 degree C, this is the highest dose area of the duct. This is 155 dpa irradiation and there is fairly little change in the transition temperature and also fairly little change in the yield stress. So, you pick the right irradiation temperature, you can maintain the properties in the material quite well.

And in summary as well what's leading to these changes in properties are at low temperature, a high density of interstitials and vacancies. At intermediate temperature there is a little bit of void swelling and some precipitation but not enough to cause significant changes in mechanical properties. And at even higher temperature, there is a little change in properties as well. But there is a class of materials that also shows a

promise as advanced radiation tolerant materials. These are called nano-structured ferritic alloys or also called oxide dispersion strengthened alloys. And in the collaboration between UC Santa Barbara and Oakridge and LANL, we have produced some of these alloys. They have a fine distribution of oxide particles, nano features, $Y_2Ti_2O_7$, and Y_2TiO_5 . It's made by mechanical alloying. And then as you develop this fine micro-structure, that leads to the particles which are shown here in the APT image that shows a fine distribution of oxide particles within the material, these serve as sinks for defects and they also help increase the strength at high temperature, giving it increased creep resistance as well as increased strength at high temperatures. So, that could help us.

You can see actually in older oxide dispersion strength and alloy MA957 that was tested in the advanced test reactor, after radiation they are only 6 dpa where I previously showed you on a ferritic martensitic steel like T91 where the uniform elongation is dropping to less than 1% in these alloys is less hardening. You can see it only it went from 1000 to 1200 mpa and the uniform elongation is still above 5% in this oxide dispersion strength in the alloy. So that may give us promise for materials for high dose applications.

So, now, I would like to shift gears a little bit and talk to you about the different reactors and their different reactor conditions and how that affects material performance. I'll talk initially about boiling water or light water reactors just as a background and then get into the different Gen IV reactor types, the thermal reactors as well as the fast reactors. So, the one that most people are familiar with that produce most of the nuclear power in the world are pressurized water reactors and boiling water reactors.

Their purpose of course is power production and the conditions. These are cooled by water so they are at fairly low temperature, 288 to 360 degree C. And they have mostly a thermal spectrum, but they do have a fast neutron spectrum as well that produces displacement damage in the material at the rate of about 2 to 4 dpa per year. Some of the issues that they run into that result in having to replace the fuel or other things are fuel clad chemical interaction, hydride formation within the cladding, and corrosion of the Zircaloy cladding as well as some issues of stress corrosion cracking or radiation assisted stress using corrosion cracking on the piping. And with the pressure vessel, as they try and push the overall age of the reactors to 40 to 60 years, it can run into aging effects for the pressure vessel as well.

And sort of in comparison, the materials that they typically look at for light water reactors are different, but there are some similarities from what we are proposing for the sodium-cooled fast reactor or the different other gas fast or very high temperature reactor for these Gen IV concept.

So, the coolant, of course is different, in the light water reactor it's water, and in the sodium fast reactor it's sodium, and then in the high temperature reactor it's helium. And the materials vary. For cladding they are proposing 9 to 12 chrome steels although most sodium fast reactors use austenitic alloys, but they are looking at changing the ferritic steels for a cladding and then in the very high temperature reactor, SiC composites are one of the prototypic alloys. And the core internals actually have some similarities. 316 stainless steel is used in sodium fast reactors and it's also used in some parts of gas fast reactors and in the high temperature areas alloy 800H needs to be used for better high temperature strength.

And then in the heat exchanger in a light water reactor, they have nickel-based super alloys, but 9 to 12 chrome stainless steel are proposed in the sodium fast reactor in alloy 617 in the high temperature gas fast reactor. So, despite the differences in operating parameters, there are some similarities across the different reactor concepts and specific areas where they are looking at more advanced alloys as well.

So now let me get into the different Gen IV reactor concepts. The very high temperature reactor is a thermal based reactor. Its purpose is more efficient power production by running at a higher temperature and it also has some inherent passive safety features.

In this case, the coolant is helium. The temperature is very high. It can be up to 950 C outlet temperature. And for that reason, it uses a solid graphite block core. And since it's a thermal based reactor, it's built up in dose is about 2 to 4 dpa per year. But the materials issues relate to improve metallic materials for the very high temperature reactor pressure vessel and as well as improvements in graphite properties. improvements in the high temperature mechanical properties in the coolant piping. Inconel-617 is the proposed material in that area. And development of better materials for the intermediate heat exchanger.

Supercritical water reactor also generally looked at as a thermal reactor, there is a fast spectrum concept as well. This is also proposed for more efficient power production. In this case you are using supercritical water coolant and so, the outlet temperature is fairly high at 550 degree C and pressure can be up to greater than 20 mpa to make it supercritical water. And also the dose systems in the thermal reactor concept are 2 to 4 dpa per year. Because of the higher temperature and the higher pressure, there are issues related to corrosion and stress corrosion cracking.

Radiolysis and water chemistry can lead to increased corrosion in the material, worrying about dimensional and micro-structural stability at these higher pressures, and embrittlement and creep resistance of cladding at the higher temperatures running at in this supercritical water

reactor. The temperature range can range from 280 up to 620 degree C and the radiation damage dose can range from 10 to 30 dpa in the thermal spectrum, but the fast spectrum can be even higher up to 100 to 150 dpa.

Now let me summarize some of the fast reactor concepts and the materials challenges there. Why look at fast reactors? Well, one of the advantages are that fast reactors can address the backend of the fuel cycle. So, we could in a fast reactor separate out the long-lived isotopes and make a new fuel to irradiate in the fast reactor to provide power with that new fuel and also reduce the need for having to store the waste in the reactor for as a long period of time as you can sort of see in this graph here. And the LWR once through cycle, you can see an increase in heavy metal mass. Irradiation will build up significantly, but if you can add a fast reactor to that, you can reduce that heavy metal mass by, in a sense, separating out those isotopes and using that as fuel in the reactor.

So what are the different fast reactor concepts? I sort of put these two together, the sodium cooled fast reactor and the lead fast reactor. In both cases, one of the purposes is high level nuclear waste transmutation and use for power production as well as using it for actinide management. The coolant can either be sodium, pure lead, or lead bismuth. And with the lead, the pure lead concept the outlet temperature is going to be higher up to even 800 degree C, whereas the sodium or lead bismuth concepts the outlet temperature is about 550 degree C. Because of the fast reactor, the total dose can build up at about 20 to 30 dpa per year, and that leads to high dose radiation effects in core, as well as especially in the lead case, you can run in the liquid metal or lead corrosion of materials.

But of the fast reactor concepts, especially the sodium fast reactor concept is the more commonly used fast reactor around the world. This is actually a picture of the FFTF Research Reactor which ran from 1980 to 1993 in Hanford, Washington.

The gas cooled fast reactor is a concept that runs at even higher temperature. It's also for high level nuclear waste transmutation, more efficient power production, and actinide management. In this case, the coolant would be helium or supercritical CO₂ and it has a fairly high 850 C outlet temperature. There are a range of different fuel options and core configurations, but because it is a fast spectrum, the dose will build up significantly at 20 to 30 dpa per year.

The materials issues, right now because of the very high temperature and the high dose fuel development is one of the major issues. You need to achieve a high power density and be able to retain fission gases at high burnup and high temperatures. One of the proposed fuels is a ceramic-

ceramic or CERCER fuel and others have coated actinide carbide, sort of kernels of fuel within the material. And there also are some alternative fuel concepts as well, fuel particles with large kernels and thin coatings and ceramic clad solid solutions.

So there is a large amount of research that needs to be done on fuel development in this area, but it could have advantages with the much higher temperature and the more efficient power production.

And then the third concept is the molten salt reactor. In this case, there is a thermal and a fast reactor concept. The purposes, high level nuclear waste transmutation is one of them. It also would give you fast reactor power. And there are different concepts. There is liquid fuel core but there also is a solid core with a molten salt coolant as a concept as well. So, the fuel could either be liquid sodium, fluorides, or chlorides. The fluoride-based fuel is the fast reactor concept. The chloride based fuel is typically the thermal reactor concept. Temperature can be up to 700 to 800 C outlet temperature. The core materials to give it the corrosion resistance are typically nickel-based alloys.

And in the case of the solid fuel, it's generally thought of as a silicon carbide-based solid fuel, but there are different options out there. But one of the advantages is it does run at low pressure as well, less than 0.5 mpa. But in the case of a fast reactor concept, the displacement damage is high, 20 to 30 dpa per year. So there are some significant materials issues. Materials compatibility that you would need to do and maybe a controlled chemistry test loop, radiation damage to the pressure vessel, and coolant piping, and materials compatibility under radiation. So, more research is needed in these areas as well.

So let me summarize the reactor operating conditions here and fusion energy is also listed here in comparison. But the fast reactors are listed up above. The lead fast and sodium fast reactor are at fuel temperature. That's the cladding temperature at 500 C to 600 C. And lifetime doses are fairly high. There you want to push the dose to 150 to 200. In some cases, there are some concepts looking at pushing doses even up to 400 or 500 dpa. In the Gen IV gas-cooled reactor, then the fuel temperature is much higher up to 2000 degree C. It's helium cooled.

Nickel super alloys are the typical structural materials with some areas where ceramic composites may be used. And structural temperatures can range from 500 out to 1200 degree C. As you then move over to the thermal reactor concepts, here is your typical light water reactor. Its coolant is water, temperature 200 or 280 to 350 degree C. The very high temperature reactor is much higher in radiation temperature, but since this is a thermal reactor the approximate doses may be about 10 dpa and the temperature is fairly high in the core.

Supercritical water reactor is UO₂ fuel, its fuel temperature of 800 to 1000 degree C and that's in the center of the fuel. But the structural temperatures are higher than the light water reactor at 300 even up to 600 degree C. And then finally the last concept, the molten salt reactor would either have fluorides or chlorides for the fuel. It could even have a solid fuel with a molten salt coolant. Fuel temperature is 700 to 800C. Nickel-based alloys are the primary structural materials, but others are being looked at. And in the fast reactor concept, those can be very high, out to 100-150 dpa.

So, and all these could lead to different performances use that I have sort of summarized here from issues for light water reactors to the thermal or the very high temperature reactor and the supercritical water reactor where helium embrittlement or creep strength can be an issue and swelling could be an issue at high temperatures. And summarizing the issues for the sodium fast reactor compared to the lead fast reactor, the issues are fairly similar except the lead fast reactor has increased corrosion from the lead coolant.

And all the way I guess to the bottom, the molten salt reactor, typical issues are corrosion of course with the coolant, helium embrittlement in higher temperatures, creep strength at higher temperatures, radiation induced segregation, transmutation, toughness, and oxidation. So, it's all summarized here. And with that I would be happy to take any questions after perhaps a few comments I believe from Patricia. So, thank you.

Berta Oates

Thank you, Dr. Maloy. If you have questions for Dr. Maloy on this presentation, go ahead and type those into the chat pod now. While those questions are coming in, we'll just take a look at the upcoming webinar presentations. We have presentation scheduled in March from Professor Abderrahim from Belgium on SCK-CEN's R&D on MYRRHA. In April, Russia BN 600 and BN 800 to be presented by Dr. Ashurko from Russia. And in May, a presentation of Proliferation Resistance of Gen IV Systems by Dr. Bari with Brookhaven National Laboratory.

Patricia Paviet

Yes, thank you, Berta and thank you so much, Stu for the excellent presentation. I would like to remind the audience of a GIF symposium in Paris is October 2018. If you want to have some information on this symposium, we put the website. The abstract deadline has been extended to the 31st of March. I encourage the professors to allow some Ph.D. students to participate to this symposium and also companies to encourage their younger staff and professionals to come and present their work. So with that, thank you very much and I think we will see the questions that you have for Stu.

END
