Advanced Gas Reactor TRISO Particle Fuel
Dr. Madeline Feltus, DOE, USA

Berta Oates
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Madeline Feltus
...is interested in what the dose would be to the public. Let's look at the parts of our program. We have fuel fabrication. We irradiate the compacts and then we do post-irradiation examination or PIES and then safety testing. Along with that we have to look at the fission product source term and how fission products are transported in the particle through the compact in the block, and we want to have modeling techniques. I want to mention how this fuel is made. We take acid-

deficient uranyl nitrate and we basically do the mixture. We make it as a stoichiometric uranium oxycarbide, and we use a solution gel, a sol–gel method, internal gelation. And what we do is we have a gelation column that has NH2OH basically in the column, and we use a needle and we make very small, tiny particles of basically jellies of fuel. We let them age in the column. We wash them with the same NH2OH. We dry. We calcine them, we reduce them, and then we sinter them.

And of course, we look at what we have. We coat them in a chemical vapor deposition reactor, a chemical reactor, and we basically go through, we compact them and then we sinter in a furnace again. In the irradiations, we have complex capsules that have to be fabricated. We have to insert them into the Idaho National Lab advanced test reactor. What's unique to our system is that we have a continuous fission product monitoring system for each of the capsules that's qualified and this just shows you our monitoring equipment. We have lots and lots of data that is quality controlled, and we have it on a special SAS-based statistics program system, MD Mass, so that we have quality assured data. We use NQA-1 2008 quality assurance and that has been accepted by our Nuclear Regulatory Commission. But you would say that’s 2008. Well, we started with the 2000 NQA-1 and always had QA as part of our program. Again, we take the particles. We can de-consolidate our compacts and then look for fission product emitting irradiated particles through an advanced gamma analyzer at Oak Ridge. We basically can heat up compacts in our furnaces.

Here's a timeline of our program. We started back in fiscal year 2004. AGR-1 was an early test of a lab scale performance. It was meant as a shakedown of the capsule design. Now, we have 72 compacts in there. We didn't just have all the same thing. We had the German-like fuel and then we actually made three variances and we tested two of them. We used one coding with argon and the other at slightly different parameters with helium as a diluent. Then, in AGR-2, we had particles made at BWXT and then we compacted them at Oak Ridge, and we have UO2 and UCO, and there was also French and South African particles in separate capsules.

We also made design to fail fuel in AGR-3/4 and that was used to look at fission product retention and how fission products moved in the graphite and the matrix. And we are currently irradiating what's known as AGR-5/6/7 where we have a qualification test of fuel totally made at BWXT, and we hope to have this completed either before or in the next cycle after our CIC which is our core internals changeout, which will be about 9 months in 2021, so this is where we are.

Alongside here, you'll see that we have Oak Ridge and BWXT fabricating the fuel. All the fuel tests are done in Idaho. And then both labs perform

the PIEs and the safety and heat up testing. And again, the whole idea is to establish US fuel capability.

This just reiterates what I've said here and shows some pictures of loose kernels and sintered kernels and how they look, before and after radiation. And again, we systematically, scientifically went from a lab scale to a totally developed industrial pilot scale. You will note that BWXT had qualification studies and we finished those in 2012. We have a completely automated compacting machine. And we can with just a single compacting machine make 5 to 10 compacts per minute. Now, obviously, we would have thousands and thousands of these in a graphite prismatic block design. And if you're making pebbles, you would have to make more than 5 or 10 per minute, but the idea was to demonstrate that we could do it. And they actually made our specimens for AGR-5/6/7 and completed them in 2006.

This just shows some of the commercial industrial scale and the overcoating and drawing was also automated as well as our hot press automation machine. And then at the end we used the sequential process without having to heat up and cool down the particles. So this way we have made better particles and compacts. Again, AGR-1 was a shakedown, AGR-2 was a performance of our test fuel that was made and coated at BWXT, and we overcoated and compacted it at Oak Ridge. AGR-3/4 is fission product transport and 5/6 is the performance of margin tests, and I'll explain that a little bit more next.

Our targeted fuel performance envelope, as I mentioned, for our program has to be larger than what was historically done for the German pebble bed designs. The Japanese have a block design. It’s a little different. They have an annular fuel. And our NGNP fits within this envelope.

And the idea here is that by having a more rigorous, more aggressive envelope, we can look at whether there are any cliffs out here. In other words, if we go too high in our packing fraction or we exceed burn up and then something happens, because our nuclear regulatory commission is always interested in the cliffs. So we bound what we’re doing for the anticipated high-temperature gas reactor designs that are being developed now. Now, AGR-2 and AGR-1 are completed, and I wanted to show you some information that we've recently presented to the NRC. And it's available in a recent licensing topical that we submitted this year to the NRC, and they are reviewing it now. I am going to focus on two of the capsules in AGR-2 capsule 5 and 6 and AGR-1, and then AGR-2 capsule 2 had the higher temperature. And you’ll notice, what happened here is that in AGR-1 we had a maximum burnup of almost 20%. This was a world record by the way. And the average temperature for the compact over the whole experiment was 1210 Celsius.

And we have a minimum, so we have to arrange – I'll explain that. And in AGR-2 we pushed it to have a high temperature capsule at 1360 maximum. You'll see that the densities are high for the power density and then our actual particle densities are higher. Now, they are a little bit different because the sizes of the kernels are a little bit different. But again, here are our burnup temperature averaged over the experiment, our power density and our fast fluence. And you'll see that our envelope exceeds this. You'll say, well, I don't see the packing fraction here. Well, we had basically 30% packing fraction and we actually pushed it even higher in our program.

Why do we use a time average value for the temperatures? Well, the population of the fuel is at different temperatures and the temperatures will decrease as we burn up the fuel. And you'll see this sort of represents what the fuel temperatures are for a certain number of particles. We actually track each compact in the set of particles here. We actually do this type of analysis to this level of detail. So about 10% of the AGR-1 fuel experienced temperatures at about 1300 C for the first 100 to 200 days and then it drops off. Okay. So you can see that we have some particles that are high. Now why would that be necessary? Because in an actual GT MHR for instance, we have different fuel loads are going through temperatures. And as we go from one cycle to the next, we're going to have reduced temperature because the enrichment is basically being reduced as we burn up the fuel. And you'll notice that our capsules envelope this range. And that was the whole idea.

This is a great picture. This shows an instrument, the line that comes out. And the AGR capsule has six capsules. It was completed in November 2009. And we've already finished our safety tests, AGR-2 had four capsules with our UCO and UO2 particles. And we had UO2 from France where they made their own compacts and South African particles, UO2 particles, that were compacted at Oak Ridge. We finished that test. We began the PIEs in 2014 and we're almost finished.

This position here are in these large B positions. We have three compacts basically spaced out in a 1-1/4-inch position with full temperature monitoring instrumentation and gas flow to monitor fission products. So it was quite a fun thing to design. We moved to the Northeast flux trap for AGR-5/6/7 and before for AGR-3/4. This flux trap is about 5 inches across and we basically have our capsules about 4-1/2 inches across, so we could get a lot more compacts done. And that's why we combined our test. If you look at our program plan, you'll see that we wanted to accelerate and get things done sooner. All right.

What's important here in this graph is that the release-to-birth as measured using krypton-85. Our US fuel was poor, but now AGR-1 and AGR-2, these are the actual capsules, have a significantly lower release-

to-birth. As I said before, the AGR-1 capsules were about 10 to minus 8. There was one that was slightly below that. Why is AGR-2 higher? Well, had higher temperatures here in our four capsules. These are ours. We do not report the South African or French information because it's proprietary. It's important to note that the success of AGR-1, we had zero particle failures out of about 300,000 during the experiment. We pushed the burnup to about 20% so we beat the German record by a factor of two, just so you can see what's happening here. And AGR-2, we had maybe zero or one exposed particle at the beginning of the irradiation. These are defects from fabrication. And there could be a possibility of a few particles failing during your irradiation. What's important here is that AGR-1 and AGR-2 showed conclusively that UCO can effectively control the carbon monoxide gas production inside the particle and prevent kernel migration, which are the concerns about using UO2 fuel to high burnups and high temperatures. What does this mean? Our AGR-1 capsule fuel failures translates into a 95 confidence failure at less than 1 E to minus 5: or 20 times better than the NGNP, our reference gas-cooled reactor design requirement at 2 E to the minus 4. Today, we can make fuel that performs as good as the German fuel at twice the burn up.

Again, I pointed to this in our earlier slide. Here's where the AGR-1 average is and here's where our burnup is. Earlier fuel experience wasn't too great and we had failures early on, but we reached a high fast fluence. That's a way we measure displacements for atom or damage in the particles at 19% FIMA. And our average was 16% and we had 610 effective full power days.

It's not good enough just to irradiate the fuel and say, ah, you didn't emit any fission products. You got to look at what's happening. So, in our PIEs, we used photographs and safety testing, advanced SEM and TEM, chemical analyses. We gamma scanned not only our compacts and the test train; we also look for the needle in the haystack. Each compact had about 3000 particles per compact and to find the ones that could have failed, we have to basically deconsolidate and look for them and search for it. We've developed different methods to do this. For instance, this is one of the particles in this pizza pie over here. That's what I call it. This is a slice across a compact, and you'll see that there are different sizes here because you are entering different particles at different heights through it. And what's important is that this white area is the silicon carbide layer, and our teams at both labs looked at this and carefully checked, as they were polishing the sample, did we have any breaches? And we actually categorized different types of breaches. Like for instance, here the buffer layer cracked and the particle oozed out, but did we have any attack of the silicon carbide layer, did we breach it? We actually looked for that. And at Oak Ridge, we have an X-ray tomography capability where we basically do an X-ray through. The particle reaction can find where the breaches are, polish and cut so that we can look

exactly where we want to look at, and then do SEM, TEM, and then very high intense TEM measurements as well. And, for instance, you could see in this one there's a little bit of palladium attack that we can actually see here on a silicon layer. This is an AGR-1 particle. It didn't fail it but you could see the attack.

AGR-3/4 again was the fission product transport test. Now, what was unique to this was we had to make designed-to-fail particles. We had to make fuel that had thin layers, missing buffers or small buffers, and we carefully took 20 of these particles where we intentionally didn't make them right. And put them in the very center of the compact. Why would you put it in the very center? Well, the MRC said, well if you don't know where you put them, then you don't know the distance how far a fission product went. So you don't know anything about the diffusion of the fission products. Well, our folks at Oak Ridge were very clever. They used basically a straw, a very thin straw, and actually carefully parked the 20 particles with a little bit of matrix and they just put it up through the metal and then they removed it with the other particles around it and compacted it and then they did radiographs. So we know where the particles are in these designed-to-fail particles. We have irradiated them in the northeast flux trap as I've said. And this flux trap is very unique. It has a nice thermal flux that represents what would happen in a gas-cooled reactor. Its prime real estate for us and we've been glad to be able to use it. And with our models, we said okay, now we have a use of designed-to-fail particles with a powerful code. When do we expect these things to start popping? Almost on cue, we started to see the first particles popping January 5th, 2012. We had the compacts. We had an inner graphite ring. We had matrix material. So when we did the PIEs, we actually could see how our fission products were going through each layer, and we know that the designed-to-fail fuel is in the center.

By the way, we used the driver fuel that we had for AGR-2 here, so we knew exactly how far fission products went. And this just sort of shows the capsules. We had 12 capsules altogether.

Now let's look at this. Did we get results that make sense here? This plot shows the log of the release to birth. Basically, it's a measure with the krypton-85, and the reciprocal of the temperature. So as the temperature goes up, you are going to basically go in the opposite direction. So we have krypton measurements. And you'll see that we have some of the historic tests, the German tests are here, the GA tests, AGR-2 you'll see – wow, we have a lot of data here.

AGR-3/4, well, yes. We have a lot of designed-to-fail particles, so okay we had a lot of temperatures and data. What does this mean? Our data is consistent with what we have seen in historical tests. The AGR fuel has a lower correlated release-to-birth as a function of inverse temperature,

so it is very, very robust. That's this average line here that was statistically checked and correlated. The AGR-3/4 result could be used by the designers of a future gas-cooled reactor to estimate what the fission gas releases would be for source term calculations. And this combined fitted line here and R/B can basically be put within a 95-confidence level here.

So if a designer uses fuel that's like what we've made, and they can demonstrate with their own proof test, they will not necessarily have to go through all the release designed-to-fail work. Okay. AGR-5/6/7 is currently in the advanced test reactor today. And again, we have five capsules. Capsule three is our margin test, our AGR-7, where we have very high-temperature fuel. The peak temperature is at 1500 C. We have compact average temperatures at 5-6 and the other capsules between 600-1400 C to bound what would happen in a gas-cooled reactor. We're seeking 6% to 18% burnup and we have 194 compacts, so it's over half a million particles to get good statistics.

Now I am just going to show you the AGR-7, this Capsule 3 here. We have fuel compacts here. We have to have a through tube for the instrumentation from the other compacts below. And the question you might ask, why does the temperature sort of change along here? Well, the flux in the advanced test reactor would peak in the middle of course with a chopped cosign and the top of the fuel, the top of the cores where the coolant comes in and it flows downward in the reactor, there's a lot of thermal analysis and neutronics analysis to determine what is happening in each one of these.

Now, high temperature safety testing of the fuel is important because we want to see that irradiated fuel will perform well in the peak accident. I mentioned this total loss of first four circulations, this coolant, this [Unclear] event. And the maximum core temperature just skirts a little bit above 1600 degrees Celsius. That's the maximum core temperature. The average is a little lower.

But you'll notice this is in hours. So it takes hours or basically days to reach this temperature, so the operators wouldn't just walk away but it's passively designed. And it's not even close to the rapid temperatures that you would see in the light water reactor event. So the fuel particles have to be designed to withstand accident conditions and retain the significant fission products. So we are going to assess these by doing dedicated testing in furnaces between 1600 C and 1800 C. We basically have the two sets of furnaces to do this in. And I just have one representative compact here. It's a 1600 C test. And what's important here is that our fuel has very low cesium released when the silicon carbide layer remains intact. There is low krypton release. Remember, krypton is a noble gas so it won't react with anything and so will go out.

The europium and strontium release, there is some but it's dominated by the amount that's in the compact matrix. It's not pure graphite. There are other materials in there, so we will see that. And silver is released and silver is not a dose that we worry about with the public, but we do have a concern for operational concerns for technicians. But again, silver can be tracked by cold traps. It's just a problem if a technician has to go into a part of the reactor coolant loop and do some work. So again, we ramp up in the temperatures and we hold it there. Now notice that we did our test for 300 hours, which is we are basically saying for instance that temperature goes up. And for this accident, we envelope it at 1600. So obviously, we are having a longer temperature than we would ever have in the reactor itself. We are going to notice in this graph that we have AGR-2 results and single particle failures for UO2 and UOC.

This looks like a big mess here, but what I want to point out here is these three lines here are the UO2 fuel, and they definitely had more than one particle released. It wasn't as robust.

There were no UCO particles that were seen in AGR-2 and AGR-1. No AGR-1 particles were seen to be failed in our safety test between 1600 C to 1700 C, so there's margin there. And there were a few failure rates when we pushed the temperatures up to 1800 C, but it was much lower than what the fuel vendor or the reactor vendor would be for the performance requirement during the accident.

Remember, that we needed to have that 10 to the minus 4. Our fractional release of cesium remained under 10 to the minus 3 for single particle releases here for our safety test. We did see some europium and higher strontium releases. But the UO2 fuel failed at the temperatures easily within 200 hours. So the UCO fuel was successfully tested. UO2 fuel would fail.

Where are we today? We basically have completed our safety tests. Our PIEs and safety test will be finished this year. We are going to complete the AGR-3/4 PIEs by 2022. We will have our irradiation finished in the ATR either by 2021 before the core changeout internals or by one cycle after that. And then we will do safety tests and heat up testing and that will be done up until 2024. And we will actually do safety tests in oxidizing atmospheres. The air and moisture test. We committed to our Nuclear Regulatory Commission to do that. We felt that it was better that we would do it than versus trying to have them do their own test. We would rather use the experimental specimens that we already irradiated, so we know it would be happening. And we have enough compacts irradiated so that we could have the statistics available to see what was happening.

We are going to compare code data calculations and see what happens to model and refine our models. And again, we're trying to support the industry so we are supporting NRC interactions on licensing. And I noted that we submitted our first topical report in May this year. And the NRC expects to have that completed by next June. They've committed to that. They've asked us a very few requests for additional information that we are responding to. The topical report is an EPRI report. The AGR team at Idaho put together a document with all the details. We had industry folks from different fuel vendors and different designers come in and critique this.

We worked that out and then EPRI published it for us and submitted it. If funding allows, we would hope to have AGR-3/4 fission product transport in another topical report to be used for source term work if possible for licensees later. And then the AGR-5/6/7 test in 2025.

Now, the question will be why did we stagger it. Well because some of the folks who had been working on AGR-1 and AGR-2 for instance and the fuel fabrication for AGR-5/6/7 will be moving on to other things. Those folks that were involved with the particular information, the particular development tasks are available now to write our first topic report and the same thing for our future topic reports.

So where are we going beyond the AGR TRISO program? Remember, we had a lab scale, very small salt gel production. We had a 2-inch coater at Oak Ridge. It was automated because we found that different technicians would change the flowrates in this coater a different times, so we automated that. And we actually study different flow gases, but we had a manual process for making the compacts. The pilot or industrial scale work at BWXT involved large 6-inch coaters, and we basically, as I mentioned before, automated the overcoating and the hot compact fabrication. Now, we didn't have 9-inch coaters where they were used in the German process, and China is using 9-inch coaters for instance, or GA used them. And why did we pick six-inch coaters? Because 9-inch coaters or anything beyond 6 inches becomes a problem with the charge of fuel and criticality safety issues. So, by having this sort of pilot scale in parallel, like let's say three or four of these, you could make a first core very efficiently.

We have TRISO fuel vendors. X-energy has pilot line work right there at Oak Ridge National Laboratory next to our AGR TRISO lab, and we have young people there that are learning from our folks, and they also are working on a X-TRISO facility funded by the Department of Energy. BWXT removed the TRISO pilot scale equipment because their other major customer needed the space. But they intend to rebuild their capability and that's been in the news. But any TRISO fuel vendor would have to have their own proof fuel test to be irradiated. AGR-8 is what we

used to call it. We didn't have NGNP materialize or go forward in funding. So a fuel vendor would have to have their own commercial proof test. They may do it in the advanced test reactor. X-energy is looking at possibly doing 8 [ph] at the ATR or in Europe. We will see.

But a fuel vendor has to do their tests under the NRC's Appendix B rule, and their results are going to be compared to our program's results. And that's why we were doing this in the first place was to provide information basically to help the licensing and the NRC had to have the data too.

I get this question all the time. Can TRISO fuel be used for other reactor designs? Of course. The molten salt cooled reactor by Kairos Power is based on the Cal-Berkeley pebble bed design that uses the graphite matrix pebbles directly, the same particles, but they are using molten salt, a FLiBe for their coolant. That's very unique. There's also a similar one at MIT that's there. But this is a company that we are supporting with other funding here at the Department of Energy.

There are fast gas reactors that are using silicon carbide or other non-graphitic matrix compacts. The French were looking at a helium fast gas design with zirc oxide coating. UC fuels in metallic claddings were being considered. General Atomic had an EM squared, an alternative design, where they actually had a TRISO- or BISO-like coating.

They also have a design with loose curls or particles. There is an encapsulated fuel form that uses TRISO fuel in SiC matrix rather than graphitic matrix that would be put in silicon carbide tubes or zircaloy cladding at Oak Ridge. This is to replace, possibly replace light water reactor fuel in our accident tolerant fuel program. And we would do this because as we saw in the Fukushima event, the zirc water reaction of the cladding in current light water reactor, BWR fuel, produces a lot of hydrogen during a severe accident. We don't have this problem with TRISO fuel with silicon carbide cladding. We do not generate H2O. And in fact because of the robust performance of the fuel, we don't have to worry about heating up and having the fuel melt like you would have in a light water reactor fuel.

We make the fuel at these temperatures. Just so you know that 300 C is the operating temperature for light water reactors, and the idea here is, excuse the units, but at 1700 degrees Fahrenheit is where the zirc water reaction takes place. Well, we make our fuel [Unclear] above that temperature, so we don't have this problem. TRISO fuel has also been looked at for fast sodium metal-cooled reactors in a dispersion-like form and TRISO fuel in a mixed oxide pellet.

I skipped a little bit fast there, there are some ideas for using refractory metals, UC and UN fuels, for space reactor designs. We have NASA using

some of our particles for their testing, and Department of Defense is also looking at this. There are designs out there that will use TRISO fuel. X-energy has their Xe-100 Reactor. They have their TRISO-X facility fabrication. They are being funded in various DOE programs right now. They have a separate agreement also with Oak Ridge under CRADA, which is an agreement to work with the laboratories at 11 million. They are now collaborating with global nuclear fuel for the DoD microreactor. The Defense Department wants a microreactor. They started their funding announcement asking for TRISO-based fuels, so we'll see. And NASA wants to use TRISO fuel possibly or basically a coated particle for space exploration. X-Energy is collaborating with Centrus for their facility design. AREVA is per the NGNP Alliance is looking at steam cycle high-temperature gas reactor design in Europe in Poland. The Department of Defense, as I mentioned, is looking at TRISO fueled mobile microreactors for strategic combat locations and possibly on their site. Some are remote. They need to have energy off the grid, and there have been different designs. The HALOS design. GA has a design. BWXT has potential applications into Department of Defense. I mentioned BWXT is a fuel fabricator. Kairos again with their pebble bed fuel that's in a salt. Urenco is looking at this for their designs and U-Battery. StarCore is working on this. General Atomics is looking at this for some of their fast gas designs and other designs. Oak Ridge has their encapsulated fuel that they're working on for replacing LWR fuel. And then of course NASA is looking at this and another company UltraSafe has various TRISO fuel forms. They are also undergoing Canadian review. The Canadian laboratory is doing reviews and might have demonstration reactors there with TRISO fuel.

I've given you some references here that you will enjoy. The International Atomic Energy Agency has various technical documents. I've listed them here that can give you information about round-robin calculations using fuel performance codes. They also have characterization round-robin exercises there. They have information about fuel fabrication in different countries, and it's basically advances in a high-temperature gas-cooled reactor technology. The Korean, US, the Chinese work is well described in this document. It's about 1000 pages, but it is like the reference textbook. And the whole idea was to try to have a bible for what gas-cooled reactor fuel and TRISO fuel history.

The US NRC has PIRT, Phenomena Identification and Ranking Table documents. There's one specifically for fuel. There's US NRC course, a part of it where we have presentations by Dr. John Hunn on fabrication, quality, and control, and you could look for that. We also have various articles for general information. I've given some of my favorite highlights. But you could look at each of these and they have further references there. And you can find these. If you need specific links, please ask.

And then I've given you some information. This is our most recent program plan. It gives information of what we were doing, where we were going. Again, this is the important one is the first topical report. We have the key results from our tests. Hans Gougar and a whole crew of us have a paper that is due out in March 2020. But you can actually get it online. It has a very nice list of the program and our progress as of this year. And you'll see different other information here.

Again, we have various journal articles that will tell you about our experiments, our PIE results, performance, and there have been high temperature reactor conferences that you can find various published information. All of our information from the advanced gas reactor program are available as Idaho external reports that will be put up on OSTI, the Office of Science and Technology Information that's headed up and held at Oak Ridge National Laboratory. You can go to Science Direct or osti.gov and look for our papers. I also have included Dr. Isabel Ben Williams and her teams' examination techniques and to explain how we've done this. And it's wonderful that we have tried all sorts of methods to do PIEs and look at silver. And the neat thing is that we have tools now that the previous TRISO fuel irradiation results could not see these things. Now, we have high-resolution TEM and focused ion beam methods, and we can see things that are on the nanoscale.

I just picked some of these. We can go from the big pizza pie down to nanometers and the backup slides also show some of that. And we're very proud that we can look at micrograph. So, with that, I would like to see if we have any questions and I really thank you for your attention and interest in TRISO fuel.

Berta Oates
Thank you Dr. Feltus. If you have questions, please go ahead and type those into the Q&A part, and we'll take those as we have time. While those questions are coming in, let's just take a quick look at the upcoming webinar presentations that we have planned. In January after the first of the year, we have a presentation scheduled on Thermal Hydraulics and Liquid Metal Fast Reactors. In February, SFR Safety Design Criteria and Safety Design Guidelines and in March, Microreactors – A Technology Option for Accelerated Innovation. There are several questions. If you can see those Dr. Feltus, I'll start with the first, a question regarding qualification. If a vendor's TRISO design includes enrichments of TRISO layer dimensions that are different than the AGR qualification programs, would the AGR qualification still be applicable if the TRISO design remains within the bounds of the reactor service envelope, possible temperature, power density? In other words, would the new design require extensive irradiation testing even if it were to be operated less aggressively relative to the AGR qualification.

Madeline Feltus
May I go to one of my backup slides?

Berta Oates
You bet. Hang on. Let me just post this question.

Madeline Feltus
And while we're getting this, the answer is it depends how far off they are. Again, this is an engineered particle. And if the kernel size is different, we have information for instance with thorium and plutonium kernels.

Can you go back for me? I only can go in one direction. Can you go back one for me?

Berta Oates
Sure. There we go.

Madeline Feltus
All right. There we go. Let me turn that off. Okay, so this is the statistical standard deviations of our fuel, and we have very tight basic standard deviations. And you will see that I've compared here the different layer thickness. Let me explain something. You have to take statistical samples as you make these and take up samples and then you are going to have a Gaussian distribution of thicknesses for your buffer, your IPyC and OPyC layers and your SiC. So here, this is the mean. But remember, the average or mean has a standard deviation. So you will see that there are different variations here. This table is actually taken from our topical report, and this is taken from one of our recent publications, and they are both in the topical report. You will see that there's a wide mean in here so that if a fabricator is within these bounds – you could interpolate, but I wouldn't extrapolate. All right? If you have, for instance, too large of an OPyC layer, that acts to constrain the particle, and the IPyC is there is compression and then the inside and are holding the SiC layer together, so you'd have to be very careful. Anisotropy is basically how random the particle is. This is the bacon factor. This is described in the tech doc. I won't go into the equations. The aspect ratio is basically how round the particles are. So the fuel vendor and fabricator who comes up with a design that has a larger kernel within these layers could do this. But if you start changing these, the way the IPyC and OPyC layer will relax and the things change as you irradiate. And we have this detail. We know what is happening. And if you were to look in the IAEA report, you will see that the particle doesn't grow but the strength of these layers changes with the fluence. Again, we are well within statistical sampling. We've got a tighter control of our process, for instance, than the German and the Japanese data or earlier US data. So, you have to have good process control.

My comment here is that the vendor would still have to do an AGR-8. They'd still have to have proof qualification tests and deviations from this, I'll call it the standard particle, would have to be demonstrated not only with irradiations but also PIEs and safety testing. So, for instance, if a vendor wanted to come in and say, well I want to come up with a different sized particle or this, why change it. As I showed in the radar plot, you can change the packing fraction. You can change the enrichment, for instance, and the packing fraction and get the reactor physics thermalization that you need. Like in light water reactors, you have different pin-to-pitch ratios. You can use the physical spacing of the particles within the matrix, the packing fraction, to get the design you want. I am going to mention the Kairos design. They need to have their pebbles to be buoyant. Now, you don't just go out and make your own particles and say oh we think we're like AGR. No, no, no. We worked with them during a game meeting and said, look, if you know how to make a pebble, for instance if they work with X-energy, if you know how to make a pebble then you can make an unfueled region in the center like a little ball. Instead of a tennis ball I would say, the outer size, you can make a little ping-pong ball of unfueled, just matrix. And then a fueled region with a packing fraction and the enrichment that you want and then rind the unfueled region and get a buoyant particle. And I did first calculations. It's going to be about an inch. And then they went back into their physics. Wow, you could do it. You can make a buoyant particle because the unfueled region is lighter than the fueled region. You've got uranium and particles in that region, so you could actually change the size of the rind for instance in your pebble bed design. In your compact, you can change the packing fraction. Now, there have been folks that've said, let's put a getter [ph] layer. In other words, with the buffer, let's see if we can put a fourth layer in there to suck up the silver for instance. If that can be done, you just have to do the test. I hope I answered that question adequately for the person who asked it. I'm glad that you asked that question.

Berta Oates
Thank you. There's another question from the same person that reads, given that TRISO form is being considered for microreactors, which were designed to load, follow more than traditional large reactors with the frequent power ramps and pose additional stress on the TRISO fuel. Are there any plans to perform tests to address power temperature ramp up/downs?

Madeline Feltus
Yes. I can't describe what the DoD wants to do, but microreactors if they do follow this – and for instance with NASA, they're going to have to go up and down. And the answer to that is you would have to do your own ramp tests as the fuel vendor to give those operating limits. Again, it could be very robust. I am sure that we won't be doing things this fast

let's say as you would do like a reactor scram in an LWR. But you would carefully be able to go up in temperature. And again, if you look at the previous slides, you will see that we do have ramps. I didn't mention it, but we actually have a histogram of a temperature for that worst-case scenario that [Unclear] where we actually are not holding temperatures constant, we actually are following that total loss of flow event and we've done that at Oak Ridge and then we are also looking at doing that at Idaho and the fuel performed well.

If you look at the elapsed time in hours, the ramp rates are on scale of maybe 5 hours. You could actually do the tests at higher ramp rates. Again, the fuel is robust. What I would not suggest doing is going out and doing like pulsing test in a trigger reactor or another device. The Japanese did that for reactivity event and basically splatted – broke the particles. Well, that was obvious. It's the total energy input into the particle determines the overpressure.

So there is a way to calculate that and that again is in the IAEA information. And the Japanese have several articles that are out there about what the power ramp effects would be and it could be done. I would hope not only would these be done for the defense reactors, but the other reactors as well. Again, load following is extremely difficult in a light water reactor because you are using control rods to do this and you get bulges in the flux when you do that. I have years of operating experience in the stone ages working at the utilities. But the nice thing here is you can do this by controlling the temperature of the helium gas and the flow rate in the gas which makes things more uniform in the core than just using control rods. We don't have control rods inside of a pebble bed, certainly not after the German crushed pebbles in one of their designs where they have control rods in the middle of it. The control rods are only used and then they use pebbles or like basically boron carbide coming into the reflector only during accident. So you can actually control the flow temperatures in the blower designs and things and have a much smoother temperature across the core.

So we aren't going to be doing those tests per se in our program because we had to do our data in the absence of a fuel design, NGNP, but we had to envelope the known areas. And if you look at the power density of the compact and correlate that to power density of the particle, that gives you a range that you can use. I hope that answered your question.

Berta Oates
Thank you. There's quite a lengthy list of questions.

Madeline Feltus
Wow. Thank you everyone.

Berta Oates
A lot of interest. The next one reads – thank you for an excellent presentation. I'm interested in how the silver-110 metastable will be handled since it migrates out of the fuel high temperatures and has a half-life of approximately 250 days. There was a mention of a cold trap in the presentation, the slide number 29. Is more analysis or report of how silver-110 metastable contamination would be addressed?

Madeline Feltus
Yes. This has been addressed in the previous Fort Saint Vrain documentation. There is also information on the NGNP design, the reference design information that Areva and others put together, the NGNP design. Basically what you can do is there's a sweet spot for when the silver comes out and the way to trap it is basically through a set of filters. Again, you're right. The half-life is long there. And so one of the best ways of handling it is to make sure that you have the off-gas system where you have the circulator and everything going in there that you basically clean up the helium gas as you go along and there you can use the cold traps to clean a portion of the helium as you go. And there are details given in that. It's also discussed somewhat in the IAEA more recent report. I have the TECDOC-1674 in front of me and some of the information on how to handle operations and some of the modelling is in there. There is a discussion of spent fuel treatment and some of the regulatory risks on this. And the IAEA TECDOCS explain some of the information on how this has been handled in the past. I would ask you do talk to either the NGNP – Farshid at NGNP, the Areva group would be good. X-energy I could refer you to Pete Pappano and the designers at X-energy as well and Kairos Power has particular ideas on how they're going to handle it.

We're not going to be doing the testing, but we basically have a way why it's going out. You know, it's one of those transition metals. Isabella's work, then Williams' work shows that palladium and silver like to go together so that's part of it is to help trap those isotopes. Okay. Next question.

Berta Oates
There is a question that says, 'Hello Madeline, I'm currently involved in the Gemini Plus project aiming at an 8QGR for an industrial heat in Poland and aim to identify initial impurities in matrix material of compacts. Any idea where to look?'

Madeline Feltus
Yes. Yes. Thank you for that question. I love it. If you look in the IAEA textbook there I mentioned, the TECDOC, there are descriptions of what has had to been looked at for making matrix material and the characteristics of the matrix. And you'll actually see, there's actually

information about the contamination levels. It becomes important, for instance, there's only so many sources of graphite available. SPL makes some. Graphtek makes some. And so what you have to do – you don't want to have to clean up the graphite, but you have to know what's in there. And there are carbon composite materials properties, there's some information about the characteristics you need to have in the matrix material. Each graphite is very unique and in the advanced reactor technology research, we have a graphite irradiation program, specifically that was part of NGNP and now continues. And they have characterized the graphite. I would suggest that you talk with Dr. Timothy Burchell at Oak Ridge National Laboratory, and he can give you a ton of information about the graphites. And more importantly, refer you to the specific IAEA text, the TECDOCS, that describe what they've been seeing. The interesting thing is that it's very dependent on what graphitic materials are being used. It's out there and it's available through IAEA and the gas reactor program.

Berta Oates
Great. It looks like about 10 questions here. We're at 8:04. I'm happy to keep going, Madeline. I know your time is valuable. If you're willing to continue going.

Madeline Feltus
I am. Yeah.

Berta Oates
Okay. Great. There's two that are kind of on similar topics. The first reads what are the plans for immobilization and disposal of TRISO fuel. And then there's a question that reads, any plans of recycling or reducing volumes of waste? And I'm not sure how much of those you want to get into but I'll go ahead and post those.

Madeline Feltus
Well, the idea here is since you have a fuel form that is encapsulated with the layers, you really wouldn't want to break it up and recycle it. And there is a discussion about the spent fuel issues in the IAEA documents 1674, it's in section – you can hear me flipping the pages. It's in chapter 11. It has experience with spent fuel treatment for the HTGRs in Germany, United Kingdom, and the US, waste concepts for what we are doing in China, Japan, South Africa and US, and advanced approaches for this.

By the way, one of the most interesting thing is that it turns out that a gas-cooled reactor can use light water reactor fuel, spent fuel, as the kernels. And you can actually deep burn the fuel to get the transuranics out. Plutonium kernel fuel has also been investigated with US funding with the Russians with the help of General Atomics. There is a Puma

Project in Europe. There was a carbo waste for fuel treatment methods. Now, you've spent all this time with the matrix material, let's say, in the blocks. You could reuse the blocks, the structural graphite, and use that as making up for your matrix material. And in that chapter 11, there is a description of how the condition is for various repository concepts. Again, we have Fort Saint Vrain fuels at Oak Ridge sitting out there. We don't have a repository yet for our LWR fuel, let alone our HTGR fuel, but there has been experience in Germany and also United Kingdom so please check out chapter 11 on the TECDOC.

Berta Oates
Great. Thank you. Can you please talk a little bit about statistical quality control techniques?

Madeline Feltus
Absolutely. What we've done in our program – let me see if I have it here. I am looking for John Hunn. You're right on it. You see this? There is a technology course, a part of the course that John Hunn gave to the NRC, and there's also a document. Can I go to the next slide, number 36, please?

Can you go to that, number 36?

Berta Oates
I'm sorry. Tell me again.

Madeline Feltus
We're on slide 35. Can you go to the next slide please? All right. Bingo. This is called stacking the duck with the NRC. The NRC has the same question. How are you going to do the quality control and the statistics? How are you going to do it? Well, the NRC needs to know how to do this. So you'll see the third bullet there. Dr. Hunn, Dr. Morris wrote basically the book on how do you do this for quality control for fabrication, and we produced this in May 2009.

This is an Oak Ridge document that is a contractor's reports, so it's a NUREG-CR out there. And what you do is you have to have statistics so you meet the 90%-95%. So that means that for instance as you coat the fuel, you're going to have to take samples from the coater. It's running. It's automatic now. You're going to go through this. But you don't want to stop the coating process, let the particles drop, unload the coater, take some out and then put it all back in. You want to do this continuously.

So this is part of what we did at Oak Ridge and BWXT is to have a method to take out samples as you go along. So what you do is you take out samples. And for instance you have to measure the density, we have a sink float method, basically a newton column with different densities. I

am sure you've seen that as a kid with temperatures. We also have methods for taking these samples, putting hundreds of particles on a plate. The team at Oak Ridge would go blind. So we have automated methods where it will calculate, okay, here's what we see as the radius, sort of like it automatically draws a circle around it. So okay, what's this radius on this side and that side so we can figure out the symmetry of the particle. And I would refer you to that document. The idea here is that as we fabricate, we have to take significant relevant samples as we go along and use the automation methods that we have to come up with these mean and Gaussian distributions. I hope that answers your question. It's easier for me to point to Dr. Hunn's document that we made for the NRC.

Berta Oates
Great. Thank you. I'm going to see if I can bundle some of these questions together. There is a question on what is the typical duration of fuel? At what rate are the pellets replaced. Does pellet replacement reduce power output?

Madeline Feltus
Okay, let me first talk about the pebbles. There we go. For instance with pebbles, you load the fuel with graphite balls only and you start by introducing pebbles at the top of the core to get a critical mass and you can recirculate the pebbles. That's what they want to do in the HTR10 in China and at Tsinghua University they actually have a mockup to look at how the fuel handling machine would be done. You could recirculate these pebbles. X-energy was looking at this at one point and Kairos for instance. And you would have a burnup monitoring device that would scan each pebble individually and it would look out and say okay what's the gamma scam on this. How far has it been depleted? So you may be able to cycle the pebbles a few times. But you have to check at each time and what you don't want to do is have hotspots in the core. And again these are running around, I don't want to say wildly. I mean they're dropping in like a gumball machine, okay, and there've been studies at MIT looking at this and Plexiglas. We funded some university projects. And Idaho National Laboratory had some folks that were looking at modeling. X-energy has some traditional VSOP method to look at streaming lines of where the pebbles would be. And of course the NRC has said, well, you don't know where any fuel pebble is at any particular point. So we are looking at ways. We've asked in a recent solicitation with universities and our small business funding opportunities, hey folks, come up with a way to inscribe or put a tag on each pebble so we keep the inventory, so we know where every pebble is, to prevent any proliferation problems, etcetera. So we'll see if someone comes up with that.

So with the pebble flow, you basically have to look at and monitor each pebble as you go through. And the fluxes is moderately flat because you have dispersed new pebbles, unburned pebbles, in with used pebbles. If you go through a once through cycle where you just put it in once, then things are moving very slowly and you would start with an intermediate enrichment and then go up to your final enrichment. Now, in a block design, you would run for 18-month or 24-month cycles and then you would have to restack the blocks. And again you would do all the core physics calculations. I think you could tell by my Ph.D. thesis that I'm a neutron jock at heart in some way and thermal hydraulics and all this. And General Atomics has done this, what you're doing is not only are you moving a column of 10 – for instance, at the NGNP have 10 blocks high, moving a column, but you have the opportunity to move them one from the bottom to the top. And you would use the same sort of, for instance, genetic algorithm or fuzzy logic or chasm or simulate, whatever you want to use for a method to put a block in a particular place. So, you would do that with the physics and then with the startup tests you would use your – when I say core instrument, you don't have instruments in the middle of a core, but in a graphite block reactor you can actually have temperature monitors and things like that. And you can actually say, this is what the flux is here at this point, to check that you loaded the fuel in properly. So with block design, you have the opportunity to refuel and move the blocks up and down actually and across the core to optimize the fuel loading. In a pebble bed, since you have flowing fuel, you can do it continuously by modeling it. Does that help, I hope.

Again, that is discussed somewhat in the TECDOCS that I've mentioned in the other ones, not the 1674 but the other one has the information about the technology on how to do that.

Berta Oates
Thank you.

Madeline Feltus
Okay. Next question.

Berta Oates
Any thoughts on long-term TRISO performance for SMR sealed cores, which could be in service up to 20 years. Are there aging effects? The same person has a question. I don't know if you can talk about any kind of enrichment levels or not.

Madeline Feltus
Okay. I can. Okay. Anybody making a reactor is going have to make it slightly less than 20%. It's like 19.8% is the high assay LEU limit. Above 20%, under IAEA standard, US standards, you have a proliferation problem. Okay? You can make a weapon out of it. I am going to make a

joke here. The folks at Los Alamos know how to make weapons out of dirt, so they believe that anything above 20% is a proliferation problem. Again, you could have a sealed reactor for a very long life. In a pebble bed, you could have it go for 3 years, but you are going to be limited to the 20%. And the same thing is true of 20% uranium. If you wanted to do a mixed oxide core, mixed oxide in the sense that the kernels, you could go longer with a prismatic design by carefully having the blocks with plutonium and uranium fuel and uranium only. And in fact, the peach bottom one reactor that General Atomics had years ago had plutonium bearing fuel in it. So you could easily go up to 10 years. I wouldn't be surprised. The issue here, however, is if you have a sealed reactor, would your regulator let you go without any surveillance? All right. Twenty years, for instance, might be pushing the limit on the structural graphite. The graphite reflector region is made out of graphite. And you would have to examine it somehow. Okay?

So, 20% uranium is the limit. If you had a deep burn reactor, some plutonium uranium mixture, you could go longer. But again you have to have some sort of surveillance of what's going on inside the reactor in the reflector in the nonfueled areas. I hope that answers your question.

Berta Oates
Great. Let's see if we can quickly get through these last others about...

Madeline Feltus
Similar one was like what level of enrichment was used in experiments and are the result sensitive to the level? Well, in AGR-1 because we are pushing it, we use 19.8%. In the reference design for AGR-2 and following, 15% is what we used. And that's adequate for the prismatic block designs. The pebble bed designs go anywhere from 8% to about 12%. So that's why we picked that enrichment. And again, you have a lot of flexibility with packing fraction up to about 40-some percent. Okay. That was a simple question.

Berta Oates
In the commentary for slide 32, you stated the fuel vendor would need to do their own tests and benchmarked by the NRC against the AGR results. Can you outline the extent of those vendor tests and likely duration?

Madeline Feltus
Okay. All right. Okay. The Feltus Fuel Fabricator for TRISO fuel. Okay. First of all, I would need to demonstrate that the fabrication method is stable. I have tight tolerances. I know what I'm doing in terms of QA, QC on the fuel. And let's say I have the Northeast flux trap. I am making compacts. I would have my fuel in the reactor for about 2-1/2 years at let's say 15 weight percent. That's about how long our irradiations last. And it would take about 2-1/2 years should do that. You have to have

the fuel cool down in the pool before you could size it before you can cut open your capsule and remove the fuel and do the PIEs.

The PIEs could take about a year. And again, you would be funded with a very specific contract with, let's say, the advanced test reactor. Now, if I wanted to vary something, before I get to that with my proof fuel I would be wise to have some scoping tests to say, okay, well instead of using a 350-micron kernel, let's say I wanted to be ridiculous and make them 100 microns. Well, then you would have enough fuel, you would have too much coating, so you wouldn't want to go in that direction. Maybe if it were plutonium, I would. But let's say I want to make a ridiculously large particle. Okay. 500 microns. Let's say I wanted to make it 700 microns. Well, then the NRC would say, you've got a lot of fission products in there. You're going to use the same size thicknesses. Prove it. So I'd have to produce some sort of modelling to say I know what I'm doing with my physics. But more importantly, I would go through and show that my statistics were there. But again, before I go to the appendix B tests, I would do some preliminary testing. For instance, X-energy is working on that and we have others who are interested in doing that sort of proof test beforehand. So it might take about five years and it might be longer depending on how far you vary, I'll say, from reference design, but we picked that enrichment to be able to handle long-lived cores. We picked the packing fraction to be high. And we've actually done studies, and this is public, we have gone up to 40% packing fraction. By the way, the theoretical limit is 66. I'll let you think about that. Like if you stack up oranges, you do it with a hexagon and you go up at the maximum per sphere packing and you do a triangular lattices two-third. So you're not going to get 90% packing fraction. I had one person, "Oh I can go 90%." I said, "No, you can't." If you have the minimum amount of matrix, it's maximum two-thirds. So there are limits on the fuel vendor's variation.

Berta Oates
Thank you for a wonderful informative summary. What are the main failure modes of the modern UCO kernel design?

Madeline Feltus
Failure modes? Okay. If we go to one of the slides I have is on UCO versus UO2 fuel.

Berta Oates
You know the number. You can scroll too. It's right before that one where I had the histogram. So it's backup slide number two, I think.

Madeline Feltus
Yeah, I didn't print these out with the numbers. Next one. There we go. All right. We have most of the failures and we've published this with the AGR-1 data and the pizza pie analysis. Lot of the failures were – I'll call it

things that we saw interesting things going into the kernel were buffers cracking. And the issue here is if the buffer adheres to the inner pyrolytic carbon as you irradiate, then cracks may form. And as we saw with some of the tomographs, the kernels will bulge out. And if that crack buffer has fission products that go right against the inner pyrolytic carbon layer, that is usually one type of failure. The other type of failure that we've seen is and they actually don't fail is the buffer tears away mostly uniformly. Okay? Again, I would refer you to the – we have the one reference that has the AGR PIE experience is one of the references that I've given you. And Dr. Petti and the whole crew put that together where we actually identified six or seven different ways it can fail and the actual number of particles that we saw this in as a representation of the whole amount. And again as you saw, we really didn't have that many failures. But when it failed, it usually failed because there was an as fabricated defect where we had a soot inclusion. The pyrolytic carbon is basically you are putting soot on the buffer layer. And you have an occlusion which basically is like a dark spot, a gold spot, and it's a bump, and so you can have cracking there.

Now, do we have fuel that is better than the UCO versus UO2? And this is why I pulled up this slide. Kernel migration, the kernel moves because of the thermal gradient. And what happens is the kernel moves and the buffer tears off. You can actually see it move within the particle. And when that happens, the kernel gets put right up against the silicon, the pyrolytic carbon layer, and you'll see palladium attack, which is not good. The carbon monoxide pressure is higher. And in the UO2 design, you'll actually see pyrolytic carbon fractures occurring. Okay? And you'll see oxide formations that attack, like cesium, palladium oxides that attack the layers. Okay? So, we've gone through this with the AGR-1 and AGR-2 PIEs where we looked at the different mechanisms and postulated what was causing them. And there's a document that's listed in the program plan where we actually do that. And let me look at my reference list. You'll see the one that was specific, it's the AGR-1 results.

Here we go. And AGR-2 results. Okay. And that's how I would check that out. Dr. Hunn and the team and also Dr. Paul Demkowicz had a combined paper, a document, on that where they actually identified the different modes, and it was like we had AB, A-type failure, B-type failure and they had a series of these all delineated out, and it was very precise. And they actually calculated and looked for a certain amount of these occurring. A specific one. Hunn, it's Demkowicz, Hunn, Petti, Morris, et al. It's on the first specific slide, it is number 3. And you'll see the key results from that is the nuclear engineering design, and I gave you the Dorry [ph] reference. And inside of that, you'll see the actual complete PIE AGR-1 document that came from Idaho.

So that gives you the big results and then it refers to the glorious details of the AGR-1 UCO. Folks, I have a photographic memory, so I can sit here and say, oh yeah, it's AB and all this and a particular diagram. But you can't read my mind, and you don't see my gestures, so maybe that's good. Okay.

Berta Oates
Can we take the maximum temperature, 1600 degrees C, as the safety limit similar to as 1204 degrees C for LWRs?

Madeline Feltus
Yes. Yes, you could. That's where you have to have a perform and that temperature was based on the prismatic block design for NGNP and the block designs. So a little bit lower in the German tests. The important part of that is if you want to look at this. There's a chapter 12 in the TECDOC-1674 that has regulatory perspectives on HTGR fuel audience safety. It was written by Stu Reuben [ph] as part of our work together as a big team. I didn't write that document. A big team wrote that document. And basically we went through and looked at the different designs of HTR10 and our tests and the GTM HR results.

I'll also let you know that there is an ANS monograph book. I am going to quote this. It's called thermal and flow design of helium-cooled reactors. It's published by the American Nuclear Society. The author is Gilbert Melese and Robert Katz. It's now published by the American Nuclear Society. You can get it online there. And it has a lot of information about the different designs, the heat transfer, the core designs historically and the process heats and things like that. But the most important thing here is that it describes what happens during accidents and the fuel temperatures and what you would see. And that would give you a good idea why the temperatures are limited. There's a very strong negative temperature coefficient, and of course the fuel has a strong negative Doppler coefficient as well. So that core response to accidents is in chapter 7. And you can see what happens on a very large 3000-megawatt HTGR plant in there and the time temperature histories are given. And so that would be a good reference if you are interested in thermal limits on the plant and what would happen during one of these events. I hope that helps.

Berta Oates
Have inconsistencies in power density particles been evaluated?

Madeline Feltus
No, the particle power densities are related to the compact densities. And again, it depends on the packing fraction that you are using. We have modeled this with [Unclear] and also our thermal analysis. And you can correlate that with burnup and what's happening and it's been shown very

truly to work. The other interesting thing is that we have to do pre-irradiation modeling and post-testing based on the detailed power history. And Blaze Colin has done that analysis for us. He's now at Kairos and William Sturbint [ph] and Bill Scourging at Idaho. So we go through that.

And by the way, we have to control the temperature and let the operating crew know what the gas flows are to control the temperature. Or what do you do if a thermocouple fails. Let's say you have no more thermocouples in one of the capsules. You don't just stop the test. We correlate that. In fact, there was an analysis where we went back and said was it adrift in a thermocouple or what really happened. So we do a lot of pre-calculations and post-calculations, and at the end of the whole irradiation run, they go through and reevaluate that again, not just at a compact level but for each capsule. So we actually go through that. It's very extensive, yes, and we haven't seen any inconsistencies yet.

Berta Oates
There are two that I think you've actually already addressed, one talking about aging consideration for long-term application beyond experimental irradiation. And then proliferation and security issues.

Madeline Feltus
Okay. Again, we are going be limited to 20%. Anyone that's making a power reactor here in United States is not going to go above 20% uranium. Period. Any IAEA, the MPT treaty adherents are going to not go above 20%. So would someone make an HEU high-enrichment reactor? You could. But it might be very inefficient for fuel utilization. You don't want to put too much fuel in the core when you don't need it. So there's no reason to go higher than 20% really unless you want to burn plutonium. Like, the Russians were looking at this for weapons-grade plutonium. And again, there is a design worked with Russia and the General Atomics folks to use plutonium, separated plutonium to burn the hell out of plutonium particles and turn the weapons into plowshares and that was the whole idea.

So again, there is a limit on how much plutonium you really want to use in a core. Proliferation – deployment of reactors is very closely watched all over the world and I can't say what a country that's not compliant with proliferation requirements would do. But I don't think they would go beyond 20 weight percent anyway. It's not efficient for a thermal gas reactor.

Berta Oates
Dr. Feltus, I can't thank you enough for your enthusiasm and passion. It really comes through when you do your presentation.

Madeline Feltus

We've been on for two hours.

Berta Oates
My goodness. This has to be a record for the number of questions and the level of interest that you have on these presentations.

Madeline Feltus
I tried to keep my slides within the 50-some minutes and we tried it out. I'm really amazed at folks' patience and duration of time, and I really appreciate it to answer questions. And again, you have my email, and if I can't answer it for you, there are other folks at the laboratories and we have staff here that are handling the graphite irradiation parts of our program and I can get you to the right person.

And again, I really appreciate folks who woke up really early in the morning to listen to this out on the West Coast. And our colleagues around the world and Poland, it must be really late. You must be doing this at dinner time and I know there are folks in Japan and China listening too because they pinged me. So again, thank you so much for your wonderful participation and questions. And again, you could submit your questions and I might even be allowed to answer them online for you. So again, thank you so much.

Berta Oates
Well, thank you everyone and have a great day.

END