The second key item is the consortium at the end of next year is to decide yes, give the blessing and go ahead with commercialization. The third thing is a continuous NRC interaction. Having an SAR by 2005 means that we interface with NRC and ACRS from beginning in a few months continuously. So when we plop the SAR on your table, you already know what it is. It's not something, good reading when you go to bed for the first time.

That way it only takes two years, 2005 and 2007. If this you see for the first time, no way you can do it in two years. We'll see each other in five years. We had that experience with AP600, so we're learning from experience. So what we want to do, this is critical, to have an interaction immediately and continuously. And achieving the deployment, of course, is the date that you saw this morning to have a U.S. generator interested by 2005. So those are the things. Next one.

So in conclusion, IRIS was designed for Generation IV. Modularity and flexibility addresses utility needs. Our first customer was DOE. At the same time we have something that is also commercial, as I went through. Enhanced safety through safety by design is a trademark of IRIS. All integral reactors have that. I think we are the one that really look and took advantage and I'm sure that what we have done will be now in other integral reactors because it just comes out of the geometry. Just comes out of that. It's physics. It's not clever design. This is physics.

It's proven LWR technology and again, I can't stress enough. We have to start testing in 2002 on selected high priority testing. Our first test will be the coupling of diversity containment just to show what you what are the predictions. That after two and a half days, you're core is still under two meters of water. I believe this is it. Thanks for your attention.

DR. KRESS: I will entertain a couple of burning questions if you have any since we're running really behind.

MR. LEITCH: The reactor vessel in the drawing looks as though it's large enough to facilitate internal control rod drives.

MR. CARELLI: Absolutely. Thanks. When I look at that geometry, it is a waste of a prime estate to have that room inside of steam generators full of control drives. The internal CRDMs are set for integral reactor. Absolutely.

MR. LEITCH: Just let me understand. The CRDMs are going to be internal? Has that decision been made?

MR. CARELLI: The CRDMs, yes. I want to have CRDMs internal. That geometry shows the CRDMs as regular CRDMs.

MR. LEITCH: Okay.

MR. CARELLI: Because the CRDMS, there are essentially two designs now. One is electromagnetic driven internal CRDMs dome by the Japanese. MHI is the one that's been testing for 10 years and again, MHI is one of our team members. The other one is hydraulically a controlled rods. And that is a solution chosen by the Argentinean, by Curum, chosen by the Chinese and actually they have a reactor in Beijing that is running right now, is operating with internal CRDMs.

So both of them and the Japanese are planning the internal CRDMs for their MRX vessel. So both of them are not a far fetch. There's been a reactor already operating or being designed. What, right now, I do not know is which one is best or better. There are two. So I have to decide which one.

MR. LEITCH: But if they're external, you haven't eliminated the rod ejection problem.

MR. CARELLI: Absolutely.

MR. LEITCH: If they're internal, you have introduced some new technology.

MR. CARELLI: Yes. You're absolutely right. There's a fine line between a deployment by 2010 and 2012 or internal CRDMs. The point again, the point is we're not starting from scratch. It has been done. There has been 10 years, 15 years work on that. What I need is about one or two years to look at critically, make a decision. At that point, we'll see how long does it take to implement. Can we make for 2012 or not? That will be the decision.

MR. LEITCH: Okay.

MR. CARELLI: But eventually IRIS is going and Curum has it, the smartest thinking about for the integral reactor is a shame to have regular rods.

MR. LEITCH: Thanks.

DR. KRESS: I think we'd better move it on now. Mr. Carelli will be available for answering other questions if you have them I think tomorrow. He'll be here tomorrow. So let's move to the next speaker which is General Atomics.

MR. PARME: My name is Larry Parme. I think most of you are new. I don't recognize you.

Perhaps a few I do. But I've been working on gas cooled reactors for about 25 years, primarily at General Atomics but I've spent time in Germany and have worked on pebble bed reactors as well, the THTR in particular, and also have worked with the Japanese in the early stages of their high temperature test reactor.

What I'd like to do over the course of the next 45 minutes, and if I can make it slightly shorter--

DR. KRESS: Please do.

MR. PARME: I will try. Next slide, please. I'll talk about the design description on the gas turbine modular helium reactor, some background to it, and then go to the key safety features, talk about the licensing approach and then the design status and deployment schedule.

As far as challenges we face in licensing, I'll point these out as we talk about the safety features and the licensing approach, and there are several challenges though I believe most of those that affect the GTMHR have already been brought up. Next slide.

The U.S. and European technology, and I don't have it listed here but I should probably also mention the Japanese as well. But primarily the U.S. and European technology gives us almost four decades of experience which the MHTGR is based.

One of the things mentioned in the earlier experimental and demonstration plants built in the U.K., Germany, the U.S. and the THTR, all of these when they were built, the vision of the future was scaling up gas cooled reactor technology in the same direction that water reactors had gone. That is, to very large, high temperature gas cooled reactors. Particularly we in the late '70s had PSARs prepared for Fulton and Delmarva. The Germans were looking in the same direction and Framatome themselves were looking in that direction.

But about that time, that is the end of the '70s going into the '80s, the same technology that had been developed out of these various reactors, we had a change in paradigm and took a second look at the design and decided that rather than scale up to -- Fulton might have been -- I believe it was about a 3,000 megawatt thermal plant and you can figure out the electric power would have been just under 40 percent efficient. Rather than go that way, we saw a different way to optimize the characteristics of the gas cooled reactor and in the U.S. we developed the modular high temperature gas cooled reactor.

This is a steam cycle plant, the same as these demonstrations plant and the same as the large HTGR would have been, but much smaller. The MHTGR design was developed to early and preliminary design in the mid '80s when we developed a preliminary safety information document and a risk assessment on the design and went for a pre-application review with NRC and also presented the design to the ACRS.

GT-MHR is an extension of that. Basically, it builds on the technology of the MHTGR. I can say there was an equivalent German design, I believe. Doctor Slabber mentioned it. The HTR module of Germany. But the U.S. design was a 350 megawatt core. What we've done is taken that, enlarged the core somewhat and replaced the steam generator with a direct cycle gas turbine, a Braten cycle loop in the other vessel. But it just builds on where we were in the mid '80s. Next slide.

You can look through your slides and you can read some of the writing yourself. I want to point out some of the main features. I guess what I'll do is you've heard about gas cooled reactors direct cycle turbines, and I'll try to point out what differences are between this and the PBMHR.

First of all, a reactor size is worth noting. It's 600 megawatts thermal. We'll talk more about that size. Electrical output is 285 megawatts. The entire primary system, that is the reactor and the turbine equipment, are all located within a below grade silo. This silo or reactor building will contain fission products or other releases, but it is not a pressure retaining structure. It is designed, if you pressurize it with your helium, to vent that helium out and, in so doing, what you do is --later when I talk briefly about some of the accidents -- is you eliminate the driving force that could exist to later carry off fission products when they do come out of this reactor during accidents.

The other thing I wanted to point out, and I have to apologize for the lack of detail here to show it, but within the silo and around the reactor is a reactor cavity cooling system. You've heard about the concept on the PVMR. The idea is similar here. The vessel is un-insulated and any heat radiates off the vessel rather than heating the concrete structures here is carried off to the environment.

On the GT-MHR the design of this system could be water or air or current reference design. It's an air-cooled system. It's naturally circulating. It operates all the time. Heat loads during normal operation are actually higher than the heat loads during accidents. But you can continuously monitor it and you know it's working normal operation.

Next slide. Could you use the transparency I have, blow this up a little bit. I can see the power point slide better. Why don't you go back to that. The colors that are sharper there helps. Taking a look at the overall design, I think the first thing you notice about the GT-MHR is the whole power conversion system is integrated into one large vessel. All of the rotating machinery is located on a single shaft. That includes the exciter, the generator, the turbine and high pressure and low pressure compressors. The shaft is for taking it apart and doing maintenance. The shaft is separable at this point below the generator so you don't have to lift the entire assembly at once. But it's on a common shaft. Surrounding the rotating machinery then is the heat exchangers.

Up above there is a compact, high efficiency recuperator and below that a pre-cooler and an inner cooler. It's an inner cooled cycle. Connecting the power conversion system to the reactor is a small vessel with an inner duct for carrying the hot gas from the reactor to the power conversion system and then returning the cold gas back to the reactor. I have a plan view of the reactor and I'll show you that in a moment, which will give you a better idea, but reactor is basically an assembly, a 10 block high core with reflector above and below built of large, hexagonal graphite block identical to throe used at the Ft. St. Vrain.

One feature that I wanted to bring up is not for decay heat removal in a safety sense but for the convenience of maintenance and operation, the GT-MHR like a steam cycle MHTGR in the '80s, has a shut down cooling system, a small circulator and heat exchanger located in this vessel that allows us to keep force circulation on the reactor core if one is doing maintenance or repair on the power conversion system.

Next slide, please. The annular core is a key design feature of the U.S. designs, and a couple of things to note. First of all, the biggest single thing for the annular core, what is it doing for us? Why do we do it? It keeps us as we have upped the power from first 200 to 250 to 350, then 450 and finally 600 megawatts, it allows us to keep the surface to volume ratio or the surface area of the vessel, the outside edge of the core. That ratio to the power develop constant. It also assures us a relatively small conduction path between the inner most heat producing rings and the vessel.

A couple of other things to note on the design is there are two sets of control rods. There's a set of start-up control rods which from here I can't read but they should be located just in the inner ring of active core. These are pulled out before operation. They're not used. They stay out. They're not used in scram. However, the normal operating control rods are located in the reflector. They're not in the active core. There's also 18 channels for reserve shut-down materials and the reserve shut-down material is just to divert shut-down mean similar to what's been used in Ft. St. Vrain and also there's a parallel in the pebble bed reactor and it's just material. It's pellets, boronated carbon that can be dropped in the core.

I want to mention a couple of other things. You'll notice there are a core barrel holding the core here. With that there's riser channels. The gas that returns to the reactor is not swept up the side of the reactor. It's not against the reactor wall. The reactor wall is exposed to it but in fact the return gas comes up this channel and is then put into the upper plenum. There is a desire to keep that away from the core. The return gas is just over 900 degrees Fahrenheit. It is a high temperature vessel. It does not use LWR materials. A nine chrome vessel. Yes, nine chrome does need to be qualified for ASME but the data is available.

Next slide. Shouldn't be any surprise here. Key to both the economics and the safety of the GT-MHR is coated particle fuel. I hope I can go through this quickly, but I'm going to go over it because it is so key to the gas-cooled reactor. You've heard about the coated particle fuel, whether it be uranium oxycarbide or UO_2 fuel laced in a buffer and then multiple layers surrounding it. I want to emphasize. These little particles are really tough things. They'll stand up to internal temperature pressures of about 2,000 PSI. You've heard about the temperature capabilities. I remind you. The case of our reactor, those particles about the size of a grain of salt or sugar are compacted with graphite pitch and then that's baked and formed into rods. The rods are placed into alternate holes in these fuel elements and then the fuel elements are stacked up to make the core.

Next slide, please. Just a couple of words on the overall cycle. I mentioned it's a gas turbine cycle. Exit temperature from the reactor is 850 degrees Centigrade. About 1,560 degrees Fahrenheit. It's quite hot. With the fuel, we're able to use these temperatures and it's quite beneficial in the Braten cycle. The temperature and the pressure is dropped by about a factor of two going through the turbine. The turbine is a 600 megawatt turbine. About 300 megawatts is going to the generator to produce electricity. Roughly 300 megawatts is going down to the turbo machinery to bring the pressure back up.

When the gas exits the turbine, it's still rather warm. About 900 degrees Fahrenheit. Rather than send that to a heat sink or try to compress it at that temperature, it's passed through the recuperator. At the recuperator we bring the temperature down to just about 250 degrees Fahrenheit. At that point it passes through a precooler where it's brought down to room temperature. At that point we can more efficiently compress the gas. You go through the first stage of compression where not only do we raise the pressure but we also heat the gas. Again, to keep the efficiency of compression down, we take the temperature back down in the intercooler, pass it through the high pressure compressor and bring it back up to the core inlet temperature of just 1,000 PSI.

At that point, we take the gas back, pick up the heat that we took out of the turbine exit gas, not waste it, and then pass it back through to the reactor. Notice that when I've come down here I've picked up the 300 megawatts that I passed down the shaft. You're looking at the heat balance here. There's 300 megawatts that's lost out the heat sinks, 300 to the compressors and the turbine.

Moving on to the safety, the next viewgraph. I wanted to emphasize again the fundamental change in design philosophy that came about for these modular reactors in the early '80s. If you look at the history of gas reactors built in the U.S., be at Peach Bottom, Ft. St. Vrain, or the large HTRs that were in the design stage, you'll notice one thing in common with all of them. They have an L over view ratio of about one. It's efficient neutronically. It's also felt to be efficient economically and keeping the vessel down and cost down.

The penalty that was being paid as these things were scaled up is you can see that the maximum core temperature and a loss of cooling, loss of coolant accident is you've got ever rising fuel temperatures to the point where Fulton peak temperatures predicted were just under 4,000 degrees Centigrade. What we've done is we scrapped the idea of trying to gain the economics in that scaling. Instead, if you look at what the modular reactor is, you see a very long thin core and then if you think about the annular core, too, you'll realize just how much the geometry has changed and, in fact, the economic penalty that could be paid.

However, what the thought is with a design where we're assured that regardless of the accident or the accident conditions that keeps the fuel below the temperatures at which you'll get gross fuel failure. The idea was to gain the economics, keep the costs of the plant down by simplifying the safety systems, the complexity of plant operation, making it simple.

Next slide. I think you may have seen the same figure cast somewhat similarly, but it's a summary of tests that have been run in primarily the U.S. and Germany. There's also some Japanese test data in my figure. What you see is all the test data on these TRISO coated particles show that for temperatures below 2,000 degrees Centigrade, there's just no experience of these things failing at those temperatures. The question was asked earlier, what about the ups and downs, the transients in normal operation? The test data have looked at Ft. St. Vrain fuel. Going up and down in temperature here has no effect on failing. Repeated cyclings at low temperatures do not affect these results.

We have established, and I notice PBMR has established similar goals. For a design goal but not actually a safety limit per se, but as a design goal, we've elected to keep the accident temperatures below 1,600 degrees Centigrade. But I want to make it clear that 1,600 degrees Centigrade is not a magic temperature. You don't go to 1,601 or 1,650 or even 1,800 degrees Centigrade and these particles to burst or anything like that. There's a time and temperature effect that occurs as you start going to higher temperatures. The time is not very long when you get up to temperatures well in excess of 2,000 degrees C. But below 2,000 degrees Centigrade, it's a time and temperature effect with degradation of the silicon carbide.

You notice the maximum peak temperature is well below that 1,600 degrees and, in fact, the average core temperature is below 1,000 degrees C.

during normal operation. Next slide.

Just summarizing where the design takes us. You can look for what I would consider to be worst case accident. You're starting with a maximum temperature of 1,200 degrees Centigrade and if you assume we lose the coolant circulation, we don't have a lot of redundancy in coolant circulation. If you lose that, there's a sudden drop in the maximum temperature and that's just the drop in the profile you get from fuel at power where there's a heat flux going out to the coolant. You had a quick drop in the maximum temperature and then there's a slow rise as the fuel heats back up. You get natural circulation within the blocks. You redistribute the heat. You eventually heat the vessel back up and you reach a point at which you just are radiating the vessel to the cavity cooling system.

If you postulate that in addition to the loss of force cooling that you also lose all the coolant, same effect occurs. First, the fuel temperature drops. Then it slowly rises and then over a period of days it continues to rise in the center, but you reach a point at which the heat is just conducted through the graphite blocks booting the reflector. There's radiation across the gaps to the core barrel in the vessel, and then that heat is radiated again to the reactor cavity cooling system.

Even if you assume that the reactor cavity cooling system fails, the effect on core temperatures is rather minimum, at least for a period of days. The vessel gets hotter, the surrounding structures get hotter, and I'm not claiming that loss of that cavity cooling system is something I'd want to deal with on a design basis event, but the fuel temperature is relatively insensitive to it as you heat up the structures that surround the vessel.

Next figure. In summary, the real safety approach on the GT-MHR is keeping the fission products right within the particles. Worse case fuel temperatures are limited by the design features of gas cooled reactor and really the properties that we've got, the low power density, the low thermal rating per module, the annular core and then passive heat removal to outside the vessel.

Finally, and something I didn't bring up. Okay. I'm sure that any number of reactors can shut down without rod motion. All I'm mentioning is that the thing has a negative temperature coefficient, like any other commercial reactor in -- I hope -- the world today. But there's something special about this. In the gas cooled reactor, there is such a large margin between the normal operating temperature of 1,000 degrees Centigrade average core temperature and the point at which the fuel starts to fail that we really have the ability to utilize that negative temperature coefficient and, in fact, if you just flip back to the preceding viewgraph, at least up until about 35 hours, at which point you start to get xenon decay, the effect of inserting the rods or not inserting the rods is not noticeable on the graph. The transients are exactly the same. The maximum temperatures. In fact, all temperatures are the same. The reactor just shuts itself down.

If you could flip two forward. I want to talk briefly about the licensing approach. I think this is something that we and PBMR share in common, a concern with the licensing approach. I tried to make the point that we've taken a fundamental change in the whole design philosophy. The large HTGR, the PSAR we are preparing for Fulton and Delmarva, the licensing at Ft. St. Vrain follow the framework that was used for water reactors and then rarely with just some exceptions and it was small.

But this approach is so different that going through the list of general design criteria or all the precedents for LWR is frustrating, it's counter-productive and there is no guarantee that it is either necessary or that it's sufficient and picks up the important things for the GT-MHR.

In the mid '80s on the MHTR, our steam cycle plant that I referred to, with DOE sponsorship, both in the design and the licensing approach, we started with a clean sheet of paper. The approach used. It says PRA. I want to make it clear. It was PRA techniques. Yes, we had a risk assessment of the plant, too. But it was using risk assessment techniques to systematically study what was important in the plant, what were the safety functions? What safety functions were needed to satisfy what goal? And reconstructed the licensing bases. This approach underwent pre-application review by the NRC and was also viewed by ACRS.

Some of the main points of it were, first of all, we looked and revisited. What are the criteria, the safety goals, top level regulatory criteria that we're striving to meet in the first place? I'll come back to that topic in a moment because it's key to be able to go through the rest of the steps.

In addition, what we did, even though this was using PRA techniques, we wanted to come up with bases that were familiar to the NRC, things like licensing bases events or design bases events, if you will, equipment safety classification, the design conditions that go with our safety equipment, and then design criteria, if you will. And I'll talk about these in a moment. But rederive them for the MHTGR. Next slide.

Top level regulatory criteria. When you go, if you're a gas cooled reactor person, when you go to the body of regulatory guidance there is, it's confusing, it's frustrating, in fact. We went back and looked at the various statements and tried to find things that really said how safe is safe enough? Somebody doesn't like the term safe enough. Choose your own, but we're trying to find some benchmarks to work for. We looked for direct statements of acceptable consequences or risk to the public or the environment. We tried to find statements that were quantifiable. We needed something that we could say. Hey, either we were that good and we were that good with margin, and it should be statements that were independent of the plant design. Don't tell me that I need an emergency core cooling system to back this up. It doesn't help me much and it doesn't mean much to my reactor.

These are not all the top level criteria that we uncovered in the '80s, but they were the limiting criteria as far as the design of the plant. I'll come back to these criteria in a couple of moments. Next slide.

Also, having gone through this evaluation of the plant and starting with our clean sheet of paper, we had gotten a handle on the safety functions that were important to the gas cooled reactor. We understood what criteria we were trying to meet and then we developed licensing basis events that were basically off normal or accident events used for demonstrating design compliance with these criteria. What we were doing is we were looking at the safety functions, we were looking a range of phenomena and a full range of frequency and trying to find what were challenges to our safety functions that would challenge staying within the regulatory criteria and then defining using our PRA entries, if you will, the types of challenges you could have and construct these events. This was done and something that would be very similar, do a water reactor. You could almost look at them after the fact as deterministic events.

After that, we collectively analyzed in

the PRA all those events to show compliance with the safety goals. The licensing basis events encompassed anticipated operational occurrences, design basis events and then something we call emergency planning basis events and we'll come to that in just a moment. Next slide.

I think this figure gives you a better idea of what I'm talking about. What we did is I have a frequency versus consequence, and this is whole body gama dose, plot and what we did is plot the various criteria we saw. We said 10 CFR Appendix I. That applies to anticipated releases so we should said it should apply to basically a frequency corresponding down to once in a plant life time. So we said once in 40 years. That was our design life time. Then we said 10 CFR 1000. Those are your design basis events. We presented arguments why the reasonable range for that is perhaps between once in a plant life time and down to 10^{-4} per year.

Also practice said that for higher frequency events rather than the full 25 rem of 10 CFR 100, some fraction of 10 CFR 100 is more important so I believe I have 10 percent of 10 CFR 100 there. Finally, for lower frequency events, we said the guiding regulations are the safety goal but you'll see something else here. The protective action guides for sheltering the public, and you'll see that plotted there and it really makes 10 CFR 100 safety goals non-issues.

We were trying in the '80s and I expect we would do the same thing in a future application to set our emergency planning zone at the exclusion area boundary. So a design criteria for us was to show that there would be no doses even for rare events, emergency planning basis events, that would exceed the protection action guides. So that's the lines here, the criteria, that's these frequency ranges we had proposed. Finally you see, using the PRA, how we had defined these events. These are not quite all the events.

The only other thing I want to point out so you understand our use of PRA and our what I would say is a risk informed decision but still putting it in an appearance that looks somewhat deterministic. You notice all these accidents here and they actually have zero dose. Those are not just the next order of magnitude down. One of the key things in the risk assessment that was done for the modular reactor was done early in preliminary design and we were trying to set our licensing basis with it, so it wasn't just a matter of quantifying those event sequences that led to releases. We assessed every phenomenological challenge of importance and defined as events not only those that had the highest releases but those that represented unique phenomenological challenges to our safety functions, and we felt that was an important part of putting the framework together that the NRC could live with.

Next slide. There's a viewgraph floating around, if anybody is interested, that goes much further than this but it didn't show up on the screen. I thought there's no point in putting it up. But for safety-related systems, looking again at what should be safety-related, we said it seems from practice that in general what's done is safety-related items in water reactors are those items that are required for your design basis events. Those items that are necessary to show that you meet 10 CFR 100. We took the same approach with this start of our safety functions and then building down further we derived those items in the GR-MHR which we claimed were safety-related and would be subject to the same rules as safety-related components in other reactor types. Next slide.

So I've been talking about something that was done in the mid '80s. How does this apply to the GR-MHR? Well, the process is absolutely generic and should be directly applicable to the GT-MHR. Our plan is to pick up where we left off before. The prior application of this to the MHTGR did not show any great sensitivity to what happened in the steam cycle, the power conversion equipment there. I wouldn't expect a lot of changes when we apply this method to the GT-MHR but there might be some differences in the licensing basis events and perhaps safety-related equipment.

Specifically, there's a potential for new initiating events because of the large and higher energy rotating equipment that we have within the primary coolant. Certainly recognize that. There's some potential for different consequences because of the higher core rating. Even though it stays within 1,600 degrees Centigrade, the same maximum temperatures the MHTGR had, it's nearly twice as large.

Finally, water ingress events in the MHTGR were a primary contributor to release. In that assessment, we would expect that our licensing basis events involving water would be very unlikely and probably be much less risk important. Next slide.

The GT-MHR is now being developed in an international program. This is being done in Russia, primarily centered in Nishni Novagrad under U.S. and Russian federation agreement and for the purpose of destroying weapons grade plutonium. Program is sponsored jointly by the U.S. DOE and Minatom, but it's also supported by Japan and -- that should be France rather than the entirety of the European Union.

The conceptual design is completed and we expect to have preliminary design complete by early 2002. I was just in St. Petersburg a couple of months back and it's quite impressive. A dollar goes a long way in Russia. There is a large staff, and they're moving along aggressively. Next slide.

The program is set to design, construct and operate a prototype module by 2009 in Thomps. We would also in Russia design, construct and license a plutonium fuel fabrication facility in Russia. The first four module plant would be up and operating by 2015 with a total plutonium consumption of 250 kilograms a year.

Just as a point of interest about GT-MHR in Russia. Fuel contains no fertile material. It's pure plutonium, weapons grade plutonium. This is not like burning plutonium with MOX or anything. There's no fertile material to make more plutonium, so it destroys it and in a burn up you get better than say on the order of 90 percent or better plutonium 239 consumption. Next slide.

Obviously, plutonium 239 and plutonium cores are not of interest here in the U.S. to our commercial program. So how does this international program relate to the commercial reactor that I'm talking about? It's basically designing a uranium fuel core in the U.S. to replace the Russian plutonium design. Next viewgraph. That's really the big picture, but there are a few other things. We are working with potential users of the technology to define the requirements appropriate to the U.S. We would anticipate doing the safety analysis and, of course, the licensing submittal would be done out of the U.S. but we would imagine doing the safety analysis ourselves, even though we may well build on analysis done by the team in Russia. Any performance assessments would also be done here in the U.S. Construction could begin with an aggressive schedule in as little as five years here in the U.S. Next slide.

I have a schedule here that hopefully you can read at your place. It doesn't look too clear up on the board. It relates the two programs. I'm going to have to move away from the microphone. I hope you can still hear me because I can't read it from there. I think the key thing to note here is the relationship between the two programs. Right now the intent is that the Russian program sets and covers the cost of design but in more than design, it especially gets much of the component testing we want done.

Construction license is looked for in Russia in about 2005 and the first prototype is built, completed 2009. If you look down at the U.S., we're talking about -- and this is the aggressive schedule-- but we've looked at it and bellevue that we can have the construction and start up by just about a year. Much of the safety analysis was already done in the early '90s. Actually a 600 megawatt core was analyzed by General Atomics in San Diego. So we're really not starting from scratch. Much of the work was done in '92, '93, '94 time frame. Putting that together and putting it together with information we would get from the Russians leading to a first plant by the end of the decade.

Particularly vague in this is the question of construction, combined operating and construction license and credifiction. The goal here is clearly to get a certification for the design. The current thinking though is the application and that's key to the program -- but that the application up front would be for a combined operating and licensing license with the eventual goal of design certification, but that is one of the things we're looking to discuss in the pre-application discussions with the Commission staff.

The other thing we're very interested in and is unique to this program and we wish to discuss with the staff is the question and possible pitfalls of bringing what was once U.S. technology back to the U.S. from Russia and one of the things we need to watch for. Clearly, the more we can bring back from the Russian Federation, the more smooth the path for this program. I will say the Russians are not off working on their own. The program is managed by DOE and they are very interested in potential market applications and are looking at, if not using, U.S. codes and standards in the design of the components and are continually asking us about U.S. safety regulations so that this could go back.

Last slide. In summary, GT-MHR is rooted in several, almost four decades of international technology and it builds directly out of the 1980s MHTGR experience. It represents an optimization of characteristics inherent to gas cooled reactors or at least high temperature gas reactors going for both high thermal efficiency with the Braten cycle, the ability of an all refractory core to go to throe kind of temperatures, but also uses those characteristics to have, I believe, simple, easily understood, assured safety. And finally, international program facilitates near-term deployment of this.

DR. KRESS: Thank you. I think I'll exercise the prerogative of the chairman and ask the first question. For light water reactors, the safety goal that you have of 5 X 10^{-7} for early fatalities. You hear statement like well, that's for light water reactors because we can live with that number because we have some idea of what the uncertainty is in the determination of it. But because those uncertainties are pretty big, we hear statements like well, we're going to not let you do that all with preventing the core damage. We're going to make you have a containment because of uncertainties. There's no quantification in my mind of what that uncertainty level is where you no longer have to have a containment. How are you going to deal with that concept in the regulatory arena?

MR. PARME: I've heard that. I've heard

those kind of questions multiple times. In the '80s, what we submitted first of all is we argued that the goal of the NRC should be to assure the safety of the public, environment if that be also the case, but the criteria for the top level regulatory criteria and going and giving me a criteria on core melt or core damage is not really telling me anything about how safe you want the public. I will admit they didn't full accept that response, but in the case of the high temperature gas cooled reactor, I'd come back in a second. Perhaps it's not such a concern if something like that were imposed on me.

In all of the accidents -- and some of the accidents I plotted up there. You'll notice all of throe things are less than a rem and typically they're on the order of tens of millirems. Some of those things include assuming that in the steam cycle plant we had lost all electric power on one module, took a break in a steam generator, lost our forced cooling, started pumping steam from one module back to the others for hours on end with nobody taking action. Those are still the kind of doses we got. There's no damage to the core.

However, I will add, we mistakenly in the mid '80s said, what do you mean by core damage?

There's no damage. The graphite will stand up to 5,000 degrees Fahrenheit or more before it starts to sublime. It won't be damaged. There's nothing here you can get temperatures like that. Well then they started redefining it as a dose over 100 millirem or something like that.

I think the argument is tell me how safe you want me to be. If Generation IV or if these newer reactors are supposed to be quantitatively safer --

DR. KRESS: If I tell you how safe I want you to be at some confidence level, will you be able to give me the uncertainties in your determinations?

MR. PARME: I can certainly try it. In fact, the submittal I will give them, the accidents we submitted to NRC on MHTGR were not quote "conventionally conservative analysis." They were run statistically and we used Monte Carlo methods to give them. I think we said what do you want? They didn't know. We gave them 95th percentile confidence on the results we give them. If you want more confidence than that, I can do it. Most of these accidents are simple enough to analyze that I can actually --

DR. KRESS: That's the problem. I don't know what confidence I want. I don't know if anybody does.

MR. PARME: I don't know but I think we can perhaps talk and work to what amounts. At this point in time, what would give you reasonable confidence? And this whole method I went through quickly but it does include -- classified events and meeting the goals. Confidence in the answers.

DR. KRESS: I'm quite pleased to see your frequency consequence curves because some of us on the ACRS think that's a good way to go, particularly when you don't have core melts.

The other question I wanted to ask you that may come up, I don't know. Chernobyl had a lot of graphite and it apparently burned. You have an air cooled cavity where you're encouraging natural convection. Is there an issue there?

MR. PARME: Let me say a couple of words. In the NRC interactions we had in the '80s, we did do some analysis of broken vessels, failed vessels, and air ingress. First of all, reactor grade graphite in the U.S., H451 for pebble bed modular reactor. I'm not sure what the grade is but typically the German graphites. They will not burn in the sense of a self-sustaining chain reaction. Coal has --

DR. POWERS: Why do you say that?

MR. PARME: I will say that exactly as

follows. Coal will burn, charcoal will burn because of its impurities. Reactor grade graphite -- and there's been tests done at Oak Ridge where an oxyacetylene torch was placed on the graphite.

DR. POWERS: It's a totally ridiculous test. You're talking of the difference between a point ignition and a homogeneous ignition.

MR. PARME: Okay. In the case where we analyzed air going into the core, and here I'll speak only of the blocks, the reaction rate is driven by temperature that is held up by decay heat. The heat generated from oxidation of the graphite was about-- and it's been 10 years -- but on the order of 10 to 20 percent of the total heat generated was -- in fact, 10 percent or less was due to oxidation. Also the reaction then becomes oxygen-limited as the air passes up the channels. We did an analysis assuming a vessel failure in that cross vessel that connects the two vessels and then assumed that the silo was open and you could get air in that. What you would get was air coming in the hot duct, going up through the core, down through the vessel and out the return duct.

We did the analysis for about 24 hours and I think we did it beyond that but, once again, I'd have to go back and look at the calculations, though it is in Appendix G, I believe it is, to the preliminary safety information document that was submitted. I think you see there's no increase in particle failures, but what you do is you are getting releases. They're pretty substantial because they're a driving force and the releases you're seeing and the doses that come with it are due to picking up the contaminants that are within the graphite. As you oxidize the graphite, there are contaminants there. They were -- I want to be careful about quoting the doses. I rather doubt that they stayed within the protection action guides for that accident. However, they were well within the limits of 10 CFR 100.

My comment on combustion was implying just primarily that the reaction is driven by decay heat. It's not as if you had a charcoal pile there. But you will oxidize. There's no question you will oxidize graphite.

Incidentally, in the large HTGR, the approach to that, if you got a break and the primary cooling system got air in the system, it's a coolant. What you do is if you've got a circulator, you turn the circulator on and you cool the core with air. Once the core temperature is down, it will not oxidize so you just run the circulator. That was the design approach for the large HTGRs. If you had a circulator running, that's how you do it. You just turn the circulator on, blow the air around and cool it off.

DR. POWERS: I'll also comment that you need to be very careful about reaction kinetics and graphite. They are catalyzed and they catalyze by the impurities he speaks of. One of the most effective catalysts that I know of, by the way, is cesium.

MR. PARME: It is effective. You're quite right about that. Fortunately, while dose-wise it's a major contributor, a fairly small amount of it that's in the graphite, but you are correct. It's a very capable catalyst.

DR. KRESS: I think with that, even though we're running considerably behind, that I'll take a 15 minute break. So please be back at 4:15.

(Off the record at 3:59 p.m. for an 18 minute break.)

DR. KRESS: Can we resume our meeting, please. I think we're on the agenda where we're going to hear a presentation on the advance liquid metal reactor ESBWR from General Electric.

I would like to note for the record that our member Peter Ford, who shortly was an employee of General Electric, has a conflict of interest on this subject and this is a formality we have to do for the record. With that noted, I'll turn it over to our next speaker.

DR. RAO: My name is Atam Rao. When I joined General Electric Company after doing my Ph.D. at Berkeley 27 years ago, they said that nuclear was going to come back in five years. Still waiting for that. I hope when it comes back there'll be nothing but a slew of ESBWR orders followed by B.S. prism as we run out of fuel with the light water reactors. Next slide, please.

ESBWR is a design that is based on the SBWR which was a 600 megawatt design and the ABWR. It basically uses a lot of the components from the ABWR. It's a natural circulation reactor. It's got a lot of the ABWR components but a lot less of them. It's got passive safety systems which were reviewed by the NRC for the SBWR program. We have done a significant optimization of the building and the structures to improve the overall economics and the construction time. It's been an eight year international design and technology program, and the goal of that program was to improve the overall performance, safety and the economics. We did stop the SBWR program because at that time we realized that it would not meet the market conditions of overall economics.

The major regulatory issues are right here on this first chart for the ESBWR. How much use can be made of the ESBWR review done by the NRC? We've done an eight year testing program. Is that enough? We've done an eight year testing program before that for the SBWR. So there's an extensive test program which has been reviewed by the NRC.

However, I'm not going to tell you how long it's going to take to license this plant. A lot of the previous speakers did tell you that. In fact, that is our biggest question at GE. We know that our experience with the last round of certifications was that it took eight years. I think the AWBR took 10 years. And the question is really how high is the hurdle and will the bar be being raised every time as you go along.

We believe for this plant design we have done all the testing. The design and the technology is complete. How long it'll take to get it through the certification hurdle is still an open question.

The next charts shows that General Electric Company had a steady program of evolving the designs, improving the reactor designs. All the actual designs started from the initial submarine reactors and we have been simplifying the design. It's interesting to see that a lot of the advanced designs that were presented earlier are either called integral design or direct cycle designs. We've had that for quite some time. Those were Generation I reactors for the boiling water reactor.

The one that I would like to mention is the ABWR. The plant is licensed, designed and operating. When it comes to regulatory challenges, we still believe that the issue of COL and ITAACS is an issue that needs to be addressed. Very generic to all of the plants, whether they come up for application in the U.S. The ABWR, we believe, hopefully will be the first in line to go through that process. The ESBWR evolved as we further simplified the ABWR. Next chart.

We also had an evolution of the buildings. There is not enough time to, like Rodney Dangerfield, I guess, if you're from California, you get little respect. You're last. You only get half the time to present each one of your reactor designs, but that's okay. They are so simple, it doesn't need much time. The ESBWR design has evolved over the years. We have evolved containment building also. The ESBWR followed from the ABWR, the SBWR and we had an earlier design of the ESBWR and now we are in the process of changing the building design.

The next chart is direct cycle, boiling water reactor. You pull the control rods, water starts boiling and turns that steam turbine. Fairly simple design. Couldn't get any simpler than that. Next chart, please.

This shows a comparison of some of the key parameters, just to put it in perspective. I have shown the SBWR in the middle there and the ESBWR on the right, the ABWR on the left. It's basically the same power level as the ABWR, like I mentioned. In fact, one of the reasons we chose that power level was we wanted to keep the components the same, the reactor vessel is the same diameter. We wanted to make sure we came up with a practical design. Our emphasis is on something that's practical that commercially viable. It is an -- circulation reactor so the fuel height is three meters compared to the 3.7 meters for a traditional boiling water reactor, and we have about 10 percent more fuel bundles, about 1,000 bundles.

We have reduced the number of control rod drives which are an expensive component of the design, and the bottom line is that last item bullet there which talks about the building size. The cubic meters from megawatt electric. Like I mentioned earlier, the ESBWR is the ABWR, just less components. And that shows up in that final number. What we have is any less systems which results in an overall smaller reactor building and containment. Next slide, please.

Like I mentioned, ESBWR is a program that's an extensive program. In fact, it's been going on. We have not talked about it much publicly. It had four elements. One was the overall requirements, design, the technology and what we were doing relative to licensing. The requirements were based on utility requirements. We've had a utility steering committee running this program for the last eight years. We have been making major changes in the overall design to improve the economics, improve the margins and improve the performance. We've had an extensive technology program with a lot of testing. We extended technology beyond that.

For the SBWR there was a major test program called TEPSS and this one NACUSP and TEMPEST is ongoing and basically the reports that were produced for the SBWR program as a result of the additional testing done in support of the ESBWR. The ongoing program, Phase 3, is a program where we are improving the overall plant margins, completing some of the testing and completing the technology reports.

Next phase would be the safety analysis report, SAR preparation and, like I mentioned earlier, the thing that we can define accurately at GE is how long it takes to produce it, how long it takes to review it. Next slide, please.

The ESBWR design is based on the SBWR. Shown on that chart is the SBWR safety analysis report. So there's a lot of paper that's been produced, a lot of design that's been done, and it's also using a lot of the ABWR components. Next chart.

It's a natural circulation reactor which is standard BWR technology. It's really hard to imagine an integral vessel where you pull the control rods out and the steam is produced at the top. It's hard to imagine anything much simpler than natural circulation BWR vessel. 7.1 meter vessel. It's about 27 meters tall. Next chart, please.

The safety systems are inside the containment. The safety systems are fairly simple. Up on the top right hand corner, the blue is what we call the water make-up system. It's 1,000 cubic meters, fairly small. You don't need much water. You've got a standard suppression pool. You can see the quenchers from the safety relief valves filling up there in green. It is raised off the base mat. It's the same size as a standard boiling water reactor.

The interesting thing about this design is that all the safety systems are inside the containment and the decay heat removal heat exchangers are setting on the top off the drive wheel above that pool up there. Next chart.

This shows what we've done over the last eight years, a comparison of the reactor and containment building of the 600 megawatt SBWR and the 1360 megawatt ESBWR. You can see that the buildings got much smaller. We have done significant optimization of the building and the systems. Next chart, please.

ESBWR design philosophy compared to the SBWR has been to increase the margins. Even though we doubled the thermal power, the overall margin, both flow -- next chart, please. What we did was we also did an extensive test program. In the handouts are actually more charts than I'm using in my presentation. There are about twice as many. They give a lot more detail on the background of the additional testing that was done.

What I mentioned earlier was the overall design philosophy has been to increase the performance margins. On this chart out here is shown some key typical parameters for the plant performance. The natural circulation flow rate, whether or not the safety relief valves open following a transient, whether minimum water level is falling in accident and what the containment pressure is following an accident. And generally the results show that ESBWR performance has been improved over the SBWR design. So even though we went up in power level, we were able to increase the margins which was a significant improvement of the overall design of the passive plant.

Next chart, please. People have been using terms like minimizing initiating events. What we've done in this basic design is that the ESBWR has no safety relief valve opening following a reactor isolation, for example. This shows the reactor pressure following a reactor isolation. Next chart.

We have adopted passive safety systems, not as a religion. Passive safety systems were adopted only if they simplified the plant design. It's interesting. The idea of the optimized plant design would be where the plant systems and buildings were set by normal operation and you got the safety systems for free. When we looked at the cost of the safety systems, we found that they are reduced so much on the ESBWR compared to the total plant design that we've essentially gotten it for free. So it seems that it'll be not possible to optimize or reduce the cost of a design like the ESBWR much further.

This shows a schematic of the safety systems, and there's not enough time to go into how the safety systems work, but let me just mention, since some of you might have heard about the SBWR. The safety systems are essentially the same as the SBWR. Here's what I call the water make-up pools which run the reactor vessel and when you depressurize the reactor vessel. These are decay heat removal condensers up on the top out here. This is for removing the decay heat following a reactor isolation. On the left side you find the passive containment cooling system, heat exchangers similar to the SBWR. The design is the same. The components are the same. We are using the same basic design philosophy as we had for the SBWR. So if someone were to ask me how long would it take for the NRC to review this, my guess is maybe a couple of weeks. As long as it takes to read the reports because there is not anything that's new and it's been backed up by additional testing. Next chart, please.

This just shows another plot of the water level following a loss of coolant accident. Again, the key thing that I want to leave you with, the message I want to leave you with is that this was the SBWR. This is the top of the active fuel. This is functional time. The water level above the top of the active fuel. The ESBWR water level is higher than that for the SBWR, so we have improved the margins so it should be easier in the review process. Next chart, please.

Extensive test program was done for the SBWR. This shows some of the test facilities. This is the depressurization valve. This was the ground water-driven cooling system test facility, and it's all real stuff. Parts of full size components were tested. Next chart, please.

The decay heat removal, similar to the SBWR design. No change in the overall philosophy. Several diverse means of decay heat removal. Next chart, please.

Again, this is where we did a lot of extensive new testing. The SBWR and ESBWR Phase I test programs are listed out here on the left side. We have completed some additional testing in the Phase 2 program which was completed in '99, and we are doing some additional testing which should be completed by the year 2002. Again, these are all confirmatory testing and we don't believe there's anything that's left out there. In fact, some of our technology partners kept asking us to define additional testing that could be done, and we just ran out of ideas on anything that could be done. So we don't think there's anyone who can think of anything else that needs to be done, but we may be wrong. Next chart, please.

This is a prototype of a vacuum breaker. I just put these charts in there to show you that this is a program where there's been hardware that's been tested. Next chart, please.

Again, there's not enough time to go over each one of these, but in your handouts there's a description of some of the test programs that we used to qualify the new features of the SBWR design. Next chart, please.

The TEPSS program was a program that was performed in Europe which was a three part program to extend to the SBWR database to the ESBWR. What we tested were some innovations that we made in the design and also the different scale for the SBWR. Next chart.

We have an ongoing design program to improve the economics of the plan further and to improve performance margins. That should warm the hearts of regulators as we are improving both the containment pressure margins and also addressing some of the issues that some of our European utilities are concerned about. But at the same time, we are fairly practical. Our overall goal is to improve the economics, and we hope to be reducing the cost of the buildings by 30 percent more while increasing the margins at the same time. Next chart, please.

We have ongoing technology programs also which should be completed by 2002 and they should provide further data for qualification of the computer codes. And finally, I wanted to leave you with just an overview just to whet your appetite for the ESBWR. It is an eight year design program where we have reduced the components in systems to further simplify the design. We have reduced the structures in buildings which we believe will simplify the design. But our goal has always been to increase the margins. As I showed you in some of the plots, we have increased the margins.

The technology program basically shows that what we've done is increase the margins over the

SBWR and we have qualified the computer codes for the incremental changes that we made on the ESBWR. Challenges for the coming year. This is the one, the BC is the biggest challenge, is how do we cross the regulatory mine field? We think we've done everything that we could possibly do that would be needed for getting this plant licensed, certified. We have the experience with the SBWR and the experience with the ABWR. We have two safety analysis reports sitting on our desk. We have done the testing. The tests were completed with our partners who were involved in the SBWR program and we can not put a number on how long it'll take, what effort it'll take, to complete certification effort.

In summary, we've completed the extensive technology program and we believe that the SBWR and ABWR experience should ease the regulatory challenges. Again, the number that I didn't have in the charts. One of the reasons for embarking on the ESBWR program was to improve the overall economics of the passive plan compared to the SBWR design and we have increased the power by a factor of two and have also improved the economies by a factor of two which is sometimes hard to do. Economies of scale don't let you do that, but there are some innovations that we've done which have allowed us to do that. So that's the ESBWR.

DR. APOSTOLAKIS: What are the most dangerous mines in the mine field that you feel we ought to be working on?

DR. RAO: Our experience on the last go round was that the fact that it was -- I'll say again -- it's a time and material effort. So there tends to be no closure when you're having NRC review of the licensing submittals, whether it's with the national labs which are consultants to the NRC staff or the NRC staff. So there is a minimum incentive for closure of some of the items. That was our experience with the SBWR in the past.

We don't think there are any technical issues that are there because we've had -- I haven't emphasized the international part of our meetings. Typically we meet twice a year and have 30 or 40 people from national labs and people from all different parts of industry. So we don't think there's any technical issues. It's just bringing the NRC staff up to the same state where we are. That's one thing.

The other question is do the people who reviewed the SBWR in the NRC staff, are they still there? I think some of them are still there. That would make it go faster. The process of someone else coming up to the same level of understanding as those who worked on it is, I think, one of the major challenges we faced in the SBWR. I remember -- I don't know whether it was Ivan Catton or someone on the ACRS. It took several years before we got people to appreciate how simple our passive containment cooling system was, for example. It was actually not a natural circulation system. It was a -- circulation system. And so if the same members of the NRC staff are not there, we might have to go through that same process again.

So it's those kind of institutional issues, I think, which will be a harder challenge for us.

DR. POWERS: Is what you're saying that you can't write this thing up so that people can understand it clearly?

DR. RAO: No. I am just saying that someone starting fresh sometimes has some preconceived notions or concepts about systems work and it does take some time for people to appreciate it. That's just human nature. I think it takes time for people to come up to speed. There is that learning curve.

DR. KRESS: I think the speaker will find that the climate at NRC now is somewhat different, and they are quite interested in closure and such things in spite of the fact that you're from California. You'll find them quite interested in not dragging out reviews and getting them done in an efficient manner. So you may be quite pleasantly surprised if you come in with an application today.

DR. RAO: You might notice this is our first coming out also. We have also sensed that there may be a change and that's why we've been working on this for quite some time and this is our first coming out on this design.

DR. KRESS: In fact, your system looks enough like reactors that NRC is used to that it almost fits into the regulatory system as it now exists and may be an easier task to get one of those licensed.

With that, I'll ask if there are any questions from the audience or from other members. Everybody is anxious to get us moving on. Good.

DR. RAO: There is one other issue that I wanted to mention that's mentioned out here. Resources. It's still our position that in the near term what we believe where the resources should be focused, you know the NRC. It's getting the plants that are already certified through the ITAACS and the COL. I mean if there was a choice of where the NRC spends its resources, that's where we would see resources being spent. This would come after that.

And after there've been 100 ABWRs built in the near term and 200 SWBRs after that, the answer to what you do when the fuel -- next chart, please. What happens when you run out of all the uranium? We have something for you for that also. That's the S-PRISM. It's a liquid metal reactor which is the next presentation. Next chart, please.

DR. APOSTOLAKIS: How many did you say? Two hundred?

DR. RAO: How many?

DR. APOSTOLAKIS: Yes. Did you just say 200? In the United States?

DR. RAO: No. I was just kidding. I don't know how many it'll take before we start running out of fuel, but this next chart addresses that question right here. I think NEI said 50. Fifty by 2020. Isn't that right?

DR. APOSTOLAKIS: I have a more serious problem. The safety goals are stated in terms of rates per year and if you have 200 units in addition to what we have now, I'm not sure that the goal should stay the same, which is now creating a new problem, I think.

DR. POWERS: George, if you doubled the number of units that we had operating, it's a factor of two. We know the safety goal so precisely the fact two makes a difference one way or another.

DR. APOSTOLAKIS: A couple of 100 I can live with but if it's a couple of hundred of this, a couple of hundred of that, as you know, pretty soon--

DR. RAO: The actual numbers, you know, I think the NEI goal was stated as 50 by 2020. In the U.S. all plants. We'd like to see them all be ours but we're realistic.

When you look at this chart of the fuel availability, it's really interesting to see why we need the fast reactor. We don't think it's needed today, but it's a design that we've worked on at General Electric for many years. Next chart, please.

Not only does it help in extending the availability of the fuel cycle, it also reduces the toxicity of the waste and the spent fuel. Next chart, please.

I'm going to go through these fast. Okay. Basically, it supports the geological repository program and it reduces the environmental and diversion risks, and that's why we think some time in the future there will be the need for a reactor like the S-PRISM. What I'm going to do is give you an overview. Next chart, please.

What I'll give you is a brief overview of the design and the safety approach. I'll also give you a little bit on the description and how it's competitive, the previous licensing interactions and the planned approach to licensing the S-PRISM. Just to put it in perspective. What's different about this liquid metal reactor compared to the ones that have seen the light of day earlier? This one, we believe, is commercially attractive. Next chart, please.

The key features of the design. It's a compact pool-type reactor with modules of about 300 megawatts electric. It's got a passive shut-down heat removal system, a passive containment cooling system. The nuclear safety envelope is limited to the nucleus team supply and located in the reactor building. We've also designed in seismic isolators so the complete nucleus steam supply system. To achieve conversion ratios less than or greater than one. Next chart, please.

The design description. Next chart, please. The power train is shown in this chart out here. What you've got is a reactor module, the steam generator, intermediate steam generator, and you've got reactor vessel auxiliary cooling systems similar to the cooling system that was mentioned for the gas-cooled reactors where you have air cooling of the reactor vessel.

The power conversion system is high grade industrial standard and it's like any of the typical plants which don't have direct cycle. Next chart, please.

Next chart shows some of the key design parameters. It's 1,000 megawatt thermal reactor module and the power block consists of two reactor modules. Its gross net electrical output is about 800 megawatts electric. And the overall plant could be put together as different modules and you could end up with about 2,200 megawatts electric, depending on the number of modules you put together.

The next chart shows a picture. On the left hand side is the reactor module out there. It's an integral design. That's a new word that I'm picking up. It's sort of fairly standard for several liquid metal reactor designs. This is the reactor module out there. This is what are the passive vessel cooling systems and this is the intermediate heat exchanger on the left side there.

The number of fuel assemblies in the next chart shows it's 138 fuel assemblies and it's fairly standard fuel for the liquid metal reactor. Moving on to the next chart, what I was going to show you was some of the numbers and the reason for considering the S-PRISM compared to some of the earlier designs of the liquid metal reactors. Next chart, please.

What it shows is that earlier designs were what we call monolithic plants and this is a modular plant. What it shows is that the cost is significantly improved, partly because of the learning curve. Skip the next chart, please. And skip the next one, also. And put that one up.

This shows a comparison of the Clints River -- reactor which is a 350 megawatt electric plant. This shows the footprint. That was followed by an ALMR plant which was 311 megawatts and, since then, GE has worked on the design we call the S-PRISM which is a 760 megawatt electric plant. What it basically shows is significantly smaller. Produces twice as much power as Clints River and it's a lot simpler.

Next chart, please. This design has had previous interactions and what I show you on the next chart is what the design and licensing history has been of this liquid metal reactor. GE PRISM program was GE funded in the years 1981 to 1984. That was followed by a DOE program of about $100 million where the PRISM design was developed and the ALMR program was one of the designs that came out of that effort. Finally, when that program was completed, GE continued developing the liquid metal reactor design and developed the S-PRISM. What we have out here is a multi-year program. For almost 20 years we've been working on this design. Spent $100 million.

And what we have is, on the next chart, the ALMR which formed the basis for the S-PRISM was reviewed by the NRC in '93-'94. There was a pre-application safety evaluation of the ALMR. It included the staff for the ACRS agreement concludes that no obvious impediment to licensing PRISM design have been identified. So what we believe is that the design out here where, again in your handouts, there's almost a 50 page handout which goes into a lot more detail of the design which there wasn't enough time to cover out here. The design is fairly well advanced and the approach for licensing the plant is shown in the next couple of charts.

Next chart, please. Land approach to

licensing the S-PRISM would be shown on the next chart which is basically a detail design, construction and prototype testing. This shows the schedule for that. It is a fairly long schedule which would take up to about 15 years, but again, as we mentioned earlier, the need for this basically arises once we start using a lot more waste or using up a lot more of the uranium.

So basically in the next chart, the key issues in a safety review would be looking at the containment, looking at the core energy potential, analysis of design basis, team generator leaks, ESA, nuclear methods, hydraulic methods, validation of the fuel database and, of course, efficient product treatment and disposal. There has been extensive experience with sodium-cooled fast reactors and -- are expected. But the key issue has always been commercial viability. We believe this design, when you look at the compactness and the overall design of this design, we don't think there's much that's not known in terms of the overall physics. The main thing is to build it, test it and test out a prototype and make sure it operates as planned. What I'd shown earlier was the overall licensing approach to getting one of these plants through the licensing process.

And the last chart is component verification and prototype testing. This shows the basic approach that would be needed for licensing this kind of a plant for testing of a prototype reactor module. Thank you.

DR. KRESS: Questions, anyone? Comments or speeches? No speeches. Seeing none, let's move on then to what might prove very interesting. Some of the NRC reactions to all this and activities they have ongoing. So I'll turn it over to whoever on NRC wants to carry the ball.

MS. GAMBERONI: I'll begin. Good afternoon. I'm Marsha Gamberoni, the acting Section Chief in the Future Licensing Organization. You might have heard the acronym FLOW in NRR. We've a panel of project managers here today from FLOW to discuss the issues in our May 1 response to the Commission's February 13 SRM. The panel members include Nannette Gilles, Tom Kenyon, Alan Rae and Eric Benner.

Our agenda this afternoon, if you can go back to the previous slide, includes discussion of the future licensing and inspection readiness easement, early site permits, the construction inspection program, status of the AP1000 review, and regulatory infrastructure issues.

The next slide shows our organization. We were established late March/early April of this year. Majority of the group is on rotational assignments, but we're currently working on permanent staffing. Our SES manager, currently Richard Barrett, reports directly to the Associate Director for Inspection and Programs, Bill Borcher.

Close near term objectives are to identify the steps needed to prepare for future licensing reviews, to determine the necessary resources and technical skills needed to perform these reviews and to identify the areas for improvement so that the reviews can be completed in a predictable time frame. I'd like to mention that we're working closely with two other organizations in the NRC, the Advanced Reactor Group in Office of Research which you'll hear from shortly, and also the Special Projects Branch in the Fuel Cycle Safety and Safeguards Division in NMSS.

I just wanted to mention two meetings that we have upcoming before I turn the presentation over to the project managers. We're meeting with the Commission on July 19 on future licensing issues, and we are also planning a workshop in late July on future licensing issues.

MR. GILLES: My name is Nannette Gilles. I'm what is commonly referred to as the FLIRA lead and FLIRA stands for the Future Licensing and Inspection Readiness Assessment. The staff was directed to perform this assessment by the Commission in their February 13th SRM, and we were asked to assess the staff's technical, licensing and inspection capabilities and identify any enhancements that would be necessary to ensure the agency would be prepared for any future licensing activities that would be ongoing.

This assessment will evaluate a full range of licensing scenarios. We will be looking at all of the processes identified under 10 CFR Part 52, the early site permit process design certification, the combined license process. We will also be looking at custom designs and also be addressing the reactivated plant licensing scenario because we do know that there has been some interest in that area.

The assessment will also look at the staff's readiness to review applications and perform inspections and specifically we are going to look at staff capabilities, and we are in the process of assessing critical skills needed to perform these actions and which areas we may be lacking resources in some of those skills. We are going to be looking at schedules, external support from this committee and from contractors and our external stakeholders, and we will be looking at the regulatory infrastructure, both at current rulemakings that are ongoing and we are be planning for possible future rulemakings that will be identified during this process. In addition, we'll be looking at regulatory guidance.

We will be making recommendations in many of these areas to the Commission, in the area of staffing, training needed. Obviously there will be training needed in some of the new technology areas. We've been making recommendations with regard to contractor supports, schedules, and again, recommendations with regard to needed rulemakings and updating for regulatory guidance documents and inspection plans. And the schedule currently is that we will complete this assessment and submit it to the Commission by September 28th of this year.

I'll turn it over to Tom Kenyon for early site permits.

MR. KENYON: My name is Tom Kenyon, and I'm working as a project manager on our early site permit efforts. Although 10 CFR Part 52 was promulgated back in 1989, the staff has not received an application for an early site permit as yet. However, talking to NEI and other industry representatives recently, we expect to receive one by mid 2002, which is why we're in the process of preparing for that eventuality.

Subpart A of 10 CFR Part 52 allows an applicant to obtain approval to build multiple classes of nuclear plants on a particular site, independent of a specific plant review. And so that allows the applicant to bank the site for future use for 10 to 20 years. This reduces the licensing uncertainty by resolving site specific issues early on in the process before the applicant has to commit large amounts of resources for the effort.

An early site permit review consists of three separate reviews. The first is site safety. Another review is in the area of environmental protection and the third is in emergency preparedness. When the staff performs a site safety review, we look at site characteristics that are specific to the site such as the seismology in the area, the hydrology, meteorology, and the population demographics. The staff looks at these site characteristics to determine whether or not any of them would preclude building a nuclear plant on the site.

Then staff also performs its environmental review. They perform it in accordance with 10 CFR Part 51 and the requirements of the National Environmental Policy Act of 1969. NEPA requires that all federal agencies use a systematic approach to consider environmental impacts of certain decision making proceedings. In this case, building a nuclear plant on the site. So the staff looks at the potential environmental impacts of constructing and operating a plant there so it can make an informed decision as to whether or not it is acceptable from an environmental standpoint to build the plant.

The staff reviews the emergency preparedness to look for potential physical impediments at the site to see if there's anything that would make it difficult or impossible to develop and implement an acceptable emergency plan. They're going to be looking at things such as the population in the area, ingress and egress routes to the site, support capabilities and facilities in the area, and any other things that could affect the emergency plan.

Staff will be working with Federal Emergency Management Agency and other federal, state and local authorities to make sure that the emergency preparedness submittal is acceptable. The staff will be interacting with the public in the form of public meetings at certain stages of our review and the public will be given the opportunity to participate in the hearing on the application.

Subpart A 10 CFR Part 52 is the regulation governing the reviews of our early site permits. We have a regulatory infrastructure in place now to do these reviews. We have regulatory guides. We have a standard review plan. We have a recently revised environmental standard review plan, and we have other guidance to support our review. We've been talking with industry representatives and other stakeholders about the upcoming applications.

We've recently had a couple of meetings with the NEI Early Site Permit Task Force to discuss regulatory issues as well as guidance questions, and we've been told, as I said earlier, that the first application is expected to come in mid 2002 with two more coming in 2003 and, despite what the slide says, there's only one expected in 2004. I apologize for the misprint. So staff right now is in the process of preparing for these expected reviews by looking at resources and skill requirements. We're going to be looking at what kind of training is necessary to make sure the staff is ready for the application review.

Next slide, please. The second topic I was going to discuss is our construction inspection program. In order to prepare for the actual construction of the plants, staff is reactivating earlier efforts that it had in revising its construction inspection program. The staff was revising the program to incorporate lessons learned from our construction inspection activities back in the 1970s and '80s and also to incorporate any changes that are needed to support inspections of plants licensed under 10 CFR Part 52.

The staff has been looking to see what needs to be done to enhance the program, and we're going to be doing such things like ensuring that there's a continuous NRC presence at the site during the construction of the plant. We're going to make sure there's a better match of inspector expertise to the construction activities that are underway and, very importantly, we're going to be making sure that the acceptance criteria is more clearly defined for what the staff is to be inspecting to.

Another issue that's going to be incorporated involves developing procedures for inspecting plant components and modules that are built at fabrication sites that are off site from the facility and then, after they're constructed, they'll be brought in and installed at the site. And of course, we're going to be developing a training program to train the next generation of nuclear inspectors.

Most of our focus has been on looking at the construction activities and inspection activities of new plants that are going to be coming down the pike over the next decade, but we recently met with Entergy Northwest to talk about the feasibility of reactivating the construction permit at their WNP-1 site in Washington state. They're in the process of performing a feasibility study that's going to be completed in August of this year, after which they're going to make a decision whether or not it's economically and practical to resume the construction activities. Of course, the staff is going to have to be prepared in the eventuality that they decide they want to come back in and resume construction and so we're going to have to have our construction inspection procedures and training programs in place in a time frame to support that kind of activity.

The last bullet is identification of an industry concern regarding the inspections test analysis and acceptance criteria that's required of plants licensed under 10 CFR Part 52. There is a concern as to whether or not the license applications need to have an ITAAC on operational program such as the quality assurance program and their security and training program. The staff is currently in the process of discussing this issue with the industry and other stakeholders and we expect to resolve this issue within the next several months.

That ends my discussion on the construction inspection program.

MR. RAE: Good afternoon, everyone. My name is Alan Rae. I'm the AP1000 project manager within the Future Licensing Group. I'm actually from Great Britain. I worked for the nuclear safety regulator in Britain which is the Nuclear Installation Dispatcher but I'm here working with NRC nine months.

In contrast to the bulk of this seminar which has been about activities for the medium and perhaps even looking forward towards the long term, the AP1000 project is a current short term project. The AP600 design certification was completed by NRC in late 1999. What we're working on at the moment in AP1000 is to look at how the design certification can be translated into potential design certification for the extended operation of the AP1000.

It was decided that this will be carried out in three phases. Phase I is about complete and was carried out under review by the staff at the end of which a letter was issued identifying six key issues that could impact the AP1000 certification. Of these, four were taken forward into Phase II. They're listed in the middle of the slide. The other two issues which was decided would not be taken further at the moment. First, the PRA that had been done for the AP600 certification. Westinghouse felt that there were no significant new issues there and they didn't need any further advice from staff before making the AP1000 application.

The second was the review of the key areas of the design certification document, as it's known. That is the case, the justification which underwrites the AP600, looking at which were the main areas that would have to be changed as this was taken forward to AP1000.

Phase II scope then was four key issues. Westinghouse is seeking further detail from the staff on the applicability of the AP600 test program to the AP1000 design, the analysis codes, the acceptability of the use of what are called design acceptance criteria. These are forward commitments given at the time of design certification which will actually be completed as part of the first of a kind or as part of a subsequent program. And lastly, the applicability of exemptions granted at the time of the certification of AP600. For that, you can read the reconciliation perhaps between the codes that existed at the time when the design was developed and the certification that was eventually given.

Of these, the major item was always going to be the AP600 analysis codes and how these were developed. Westinghouse presented a report on this code development supplied to NRC in May. There's some work been done by staff getting themselves familiarized with the issues within that report. There's a meeting later on this week at which Westinghouse will present the contents of that code report and hopefully dialogue on how we're going to get the regulator assurance that's required to complete this stage of the review.

Phase III of the AP600 review will be a conventional design certification and it's expected that Westinghouse will come forward with that in 2002. Thank you.

MR. BENNER: And lastly, I'm Eric Benner, the Regulatory Infrastructure lead for Future

Licensing Organization. My blanket statement on this is what I'm about to discuss are known to-dos. These are things that were either already being worked before the creation of FLOW or have been brought to our attention subsequent to the creation of FLOW. The readiness assessment being performed by Ms. Gilles and her group is doing a more thorough scrub of the regulations to see what changes would be necessary to support future licensing activities. So we'll have a more detailed picture when that's complete.

The first item that we have going on is a rulemaking to update 10 CFR Part 52. You've heard a lot of references to 10 CFR Part 52. That was put in place as an alternative licensing method and it discusses combined licenses whereas the previous licensing contained in Part 50 dealt with the construction permit and operating license. 10 CFR Part 52 discusses a combined license which really wraps those two items together. It also makes provisions for early site permits, which Mr. Kenyon spoke of, and design ,certifications which is basically when you take a design and certify it not to license to operate but for someone to just manufacture so that someone else could license it at a later time.

This rulemaking is basically to clean up some loose ends after Part 52 is issued. After three design certifications were done, there were some lessons learned from that. That'll be incorporated. There'll be some deletion from Part 50 of repetitive appendices now that Part 52 is established. There will also be some incorporation of general provisions, licensing provisions, under Part 52 from part 50 that again, on a look back, it seemed like the general provision should carry forward.

Basically, where we're at now is there was a preliminary letter that went out some time ago asking for some comments on this, and the staff intends to issue a proposed rule package in September of this year.

There are also two other rulemakings ongoing. They both involve some of the NRC's environmental regulations. The first is a rulemaking on alternative site reviews. Basically, 10 CFR Part 51 is how the NRC incorporates the National Environmental Policy Act. One of the keystones of that act is the assessment of alternatives to any action that's being taken. The NRC has narrowed it down to look at one of the alternatives that should be looked at is, hey, you're planning on putting this power plant at this site. What alternative sites should you look at?

In the past, that was a little easier task because you had utilities that had distinct service areas. So the alternative sites could reasonably be limited to that utility service area. Now with both deregulation and consolidation, you get to a point where you could look at alternative sites much more broadly. So the staff is currently looking at how that should be dealt with. That's very preliminary at this point. We're anticipating an initiation of rulemaking mid fiscal year 2002.

The last rulemaking is environmental regulations. Tables S3 and S4 in Part 51. What these tables basically list are ramifications of the nuclear fuel cycle. It lists things like average effluence for reactor, any land and resource uses, and there are some comparisons for each of these aspects to coal power plants.

Part of the changes that have to be done are because all those tables, all the data in those tables are referenced solely to light water reactors. So obviously you've heard today about a lot of lot on light water reactor technologies, so there could be considerable work to be done there.

There's also going to need to be an

assessment done of the fact that some of these new technologies use higher enrichment uranium, so all these tables do have some bounding uranium enrichment that it deals with. Again, at this point, that's preliminary activity and, again, I think we're talking about initiation of rulemaking some time next year. Next slide, please.

Also at this time, we're not talking about implementing any of this by rule change, but instead some of it deals with interpretations of rules are the NRC's financial-related regulations, specifically anti-trust, decommissioning and modular plant requirements. That's specifically to Price-Anderson. That last one, basically the Price-Anderson Act talks about retroactive liability and it imposes a financial burden per facility and if you look at the modular plant design, say you have 100 megawatt module, currently if you just looked at how our regulations are structured, we equate a reactor to a facility. So you could have 100 megawatt module paying the same amount as 1,000 megawatt light water reactor. There is some assessment going on now as to what is truly fair, and I can't presuppose what the answer will be there, but we understand there are some concerns.

The anti-trust and the decommissioning

funding requirements. Some of throe questions again come about because of deregulation of the electric power industry. There's assessments as too -- again, in the old days, the utility owned the plant, owned the transmission lines and what not, so there were more concerns about anti-trust. Now licensees are coming and talking about making argument. The merchant plant arguments say, hey, we're building one of these plants to provide supply in the competitive market. There should be no anti-trust issues there when you're looking at that.

Some future activities that we have earmarked, and I understand that some of this is going to change. The Nuclear Energy Institute has talked about a petition for rulemaking for a generic regulatory framework performance-based, risk-informed, a pretty large scope activity. I understand now that the mechanism for that may change from a petition for rulemaking just because there are restrictions on the interfaces that the NRC can have with petitioners but suffice it to say that that would be a large scale activity as to how to risk inform the licensing process.

The last thing on my slides is really just a mechanistic thing. There's been a lot of talk now about schedules and regulatory hurdles and mine fields, I believe was the word. We understand that rulemaking by its very nature can be a long process. Some of these advance technologies don't fall nicely into our current licensing schemes because they are all geared towards light water reactors. The beauty of the design certification process is long-term, that the design gets incorporated into 10 CFR Part 52. That's a very clean, open process, but it is time consuming. It does take some time.

In the short term, we have licensed non-light water reactor technology in the past. You've heard of some of the examples. Fort St. Vrain and what not. Basically the mechanism would be to use the current regulations and for those areas where regulation intent may not apply, there would be an exemption granted if the argument was made and in those areas where the regulations may not be sufficient, then the NRC can use license conditions to incorporate other requirements. So that's just kind of plug for where we're at. That's the end of my presentation.

MS. GAMBERONI: That concludes our presentation.

DR. KRESS: Okay. I think we'll entertain questions on this part of the presentation. M R . LEITCH: Question about early site permits. Where a site was approved for multiple reactors and only one was built, does that other unit have to go for an early site permit or is that site for a potential second unit considered banked?

MR. KENYON: I'm not sure. Are you saying under the old Part 50 licensing?

MR. LEITCH: Yes. In other words, they had approval to build two units but only built one.

MR. KENYON: Under Part 50.

MR. LEITCH: Yes.

MR. KENYON: That's not really banked under the Part 52 rule. What's occurred is that when we license that plant, say we approved it for two nuclear plants, that was licensed to a specific plant design. I'll just pick on a BWR design, for instance. Therefore, although the construction permit and the license that they had would only allow them to build the same plant on the site. So if they wanted to build an ABWR there, they would have to come in for a different permit.

MR. LEITCH: My question really was if they wanted to resume their original intent.

MR. KENYON: To build the older design?

MR. LEITCH: Yes.

MR. KENYON: I'll defer to Mr. Jerry Wilson who's our PAR 52 expert.

MR. LEITCH: I was specifically thinking, I guess, of I think it's Perry.

MR. WILSON: Jerry Wilson, NRR staff. Your question gets to whether or not the original construction permit is still in effect. Assuming that it was in effect, they could use that construction permit and build another one of that design, although the designs we're talking about are quite old at this point and I'm not sure that anyone is interested in doing that.

MR. LEITCH: Okay. Thank you.

DR. KRESS: Okay. Let's move on to the presentation from NRC Research. We'll do a little musical chairs here, I guess.

MR. FLACK: My name is John Flack. I am the Acting Branch Chief in the Office of Research, Regulatory Effectiveness and Human Factors Branch. This branch will become the focal point of advanced reactor activities in the Office Research. We have a small group.

DR. APOSTOLAKIS: And human factors, you said?

MR. FLACK: Yes, human factors. Did I miss that one? We're in the process of transitioning to pick up the advanced reactor work, so what I'll do is I'll briefly go over the activities that are ongoing now in the office and the more specific activities with respect to the pebble bed Stu Rubin will cover.

Historically, the office has been involved in pre-application reviews that go back to the 1980s. This was on the MHTGR, PRISM, SAFER. In many ways, it enhanced the understanding of the concepts and really set the stage for licensing applications. There's really, I count up about five important areas and features of the pre-application review and the outputs. First, it all starts with promoting regulatory effectiveness by identifying early safety, policy, licensing issues, and then the basis for the follow-on resolution of those issues.

It also provides important feedback to the Commission and the stakeholders involved in entertaining an application for the advanced design. It also helps to generate Commission guidance on regulatory approaches that differ, sometimes substantially, from light water reactors. It identifies infrastructure needs, in-house expertise, and it also allows us to hold workshops and interface with the ACRS, which is one of the important items on our list. Again, the Advanced Reactor Group that's being formed in the Office of Research is in the Division of Safety Analysis and Regulatory Effectiveness.

On the next chart. Advanced reactors have greater reliance on new technology and that indicates the needs for new safety licensing criteria as we move toward risk-informed performance base initiatives. The pre-applications give us the introduction, you may say, to entertaining these new ideas. In an EDO memo issued in November, 2000 the Commission articulated the responsibilities of these advanced reactor reviews and in the next three bullets that I have on the viewgraph, NRR has the lead with research support for the light water reactor, advanced reactor pre-application initiatives, NMSS with the fuel cycle transport and safeguards, and Research has the lead for the non-light water reactor, advanced reactor, pre-application initiatives with longer range new technology initiatives that would essentially establish the infrastructure for the follow-on licensing application.

The memo also identified Research as

having the lead on the South Africa PBMR in coordination with NRR to plan and implement work in that area. Recent industry requests for pre-applications are listed there. Westinghouse with the AP1000 last year 5-4-00, Exelon with the pebble bed came in December. The next two, General Atomics GT-MHR. We've met with them and essentially responded to them leaving the door open for follow-on discussions on pre-applications. And then there's the Westinghouse IRIS. We had a meeting with them on 4-6 of this year.

In addition to throe pre-application interactions, there is the NEI risk informed framework for advanced reactor licensings which we are waiting the review. Next chart, please.

I'll briefly go through the PBMR. Stu will focus more on the details of that review, but basically we're engaged with Exelon on that review. There was a plan developed that was put out in SECY-01-007 but at the moment I'm not aware that it's publicly available, but it will be any day now. Pre-application work is under way and with again the objective identifying issues, infrastructure needs and framework for the PBMR licensing.

The GT-MHR. Again, we just met with them and really we're just saying that the door is open. WE're waiting for them to take the next step on that. We're thinking about time frame 2002 for initiating a pre-application review. Next slide, please.

IRIS is similar. This was a design developed under DOE, an area program which I understand you heard about earlier today. We met with them on 5-7-01 and again we are expecting a pre-application review, possibly in next fiscal year.

Generation IV is an area where we've been observing. It's an international activity coordinated by DOE. It's a longer term effort. We're thinking of designs out to 30 years, but basically we've just been gathering information and passing that on to the Commission and staff to keep abreast of those ongoing activities.

And the last activity that we're involved in or anticipating being involved in is the NEI developing proposal on the generic framework, of course, that leading to the need for NRC to establish an effective and efficient risk-informed and performance-based licensing framework.

DR. APOSTOLAKIS: John, I'm a bit confused. If someone comes to you using Part 52, is there anything there that says that you need the risk-informed performance-based system?

MR. FLACK: There's nothing in Part 52 that says that we need to have a risk-informed performance-based licensing approach.

DR. APOSTOLAKIS: So they could approach the licensing issue without using risk information. Could they?

MR. FLACK: Yes, I would expect that would be the case.

DR. APOSTOLAKIS: Is there anything that gives you the authority to request risk information?

MR. FLACK: Other than the requirements on the PRA. I think Jerry Wilson might be the one to answer questions regarding the PRA under Part 52 requirements there.

MR. WILSON: Jerry Wilson, NRR. The Part 52 licensing process is just that. It's a licensing process, and so it references back to parts 20, 50, 70 and 100 for the actual safety requirements. So whether or not those safety requirements remain as they are or change as a result of some risk-informed process, it will use whatever is the requirement that's currently in place.

DR. APOSTOLAKIS: I mean the slide said need for NRC to establish an effective and efficient risk-informed licensing framework.

MR. FLACK: That's an internal processing.

DR. APOSTOLAKIS: What if the industry doesn't want to use risk information? What if they just want to use existing regulations with exemptions or changes and maybe they feel that going to a risk-informed system adds an impediment because we have to understand it and do it. It's new. And try to go with the existing system and maybe a PRA would be an assessment at the end if you guys request it but maybe it will be a good idea not to bring it up at all. Why is that the need?

MR. FLACK: I think it would be to their advantage to come in that way. Stu.

MR. RUBIN: Stu Rubin, Office of Research. I would point out that the Commission's advanced reactor policy statement that was issued in the '80s does allow, if not encourages, applicants or pre-applicants for advanced reactor designs to submit along with their designs proposals for new kinds of regulatory frameworks, frameworks that are less prescriptive than the current basis of looking to Part 50 and looking at exemptions.

So it is an option on the part of any applicant to go with the existing framework or to propose a new approach to licensing for their design. So it is very much an option for them, and there's a decision that needs to be made whether or not it's an attractive option to try to plow new ground to develop a new framework or go with existing framework which we all know has significant burdens associated with it.

DR. POWERS: George, it seems to me that the Commission has made it clear that when the staff thinks they want information, they can ask for risk information.

DR. APOSTOLAKIS: I think they have to give some argument though that issues of adequate protection are involved. Isn't that correct?

DR. POWERS: No. They have to give an indication that there's substantial risk associated with the idea, whatever concept is put forward.

DR. APOSTOLAKIS: Which comes close to touching on adequate protection.

DR. POWERS: Shouldn't be terribly difficult to come up with those ideas. It's an interesting thing because risk has been notably absent in our discussions today.

DR. APOSTOLAKIS: Yes. I mean we keep talking about risk-informing the regulations and yet major regulatory decisions right now are being made without risk information. For example, license renewal. I believe the power operators do not use this information.

DR. POWERS: Within this context of advanced reactor codes. I guess it surprised me how little risk information has seemed to be involved in those designs.

MR. FLACK: You seem to support the bullet, the need for it.

DR. APOSTOLAKIS: No. I just was wondering whether there's a real need. I think there is a need.

MR. SHACK: This relates to the NEI proposal. NEI sees the need. You can ask them why they see a need.

DR. APOSTOLAKIS: But again, the NEI may propose an option.

MR. FLACK: Moving right along, the last slide that I am about to present is the --

DR. APOSTOLAKIS: John, before we go on. How hard do you think it would be to satisfy this need? Are we talking about a 10 year effort or are we talking about maybe a year or two?

MR. FLACK: I think the need is to improve it. Where you stop, I don't see there's any clear cut-off where we'd have enough of it. I think it's something that continues to grow and you develop. Maybe more sometimes than another but I don't see any specific cut-off on it.

DR. APOSTOLAKIS: Well, it depends. I mean if one wants to get rid of their notion of design basis accidents and use instead the PRA, then it's not obvious how one would do that. So that would a very ambitious task.

MR. FLACK: We use the PRA to pick the design basis.

DR. APOSTOLAKIS: Well, that, too. That would be -- okay. Fine. Thank you.

MR. FLACK: The last slide which I'll present is on significant technology issues, and obviously we could spend a lot of time looking at these issues one by one. I just put it up to get a feel for the kinds of areas that are highlighted and need for NRC to really understand with confidence the advanced reactor designs when pushing forth these regulatory changes.

If there's no other questions, I'll turn it over --

DR. APOSTOLAKIS: No, there is one. We heard today from several speakers, I think, that they're trying to reduce involvement of the humans. Do you think that the human performance issue will be as important here as the current reactors?

MR. FLACK: I've discussed this at length. I don't know whether we can say it's going to be less important. I mean it's going to be a different environment which that human operates in, and one has to understand that environment and what's changing in that environment. So it's something that one has to look at very carefully. So it's hard to say.

DR. POWERS: It seems to me that the change is really entertaining and in the direction that's most difficult for us because as they design the plants to be less and less dependent on the human operator intervening, seems to me we become more and more worried about the fact that the operators are not going to sit there and do nothing and they will intervene and the potential for them to intervene incorrectly in a system that's designed to operate with rather minor low head forces operating on it.

So you get into the problem of errors of commission that we are most incapable of addressing. It's a subtle problem.

MR. FLACK: Yes. The environment changes and you don't really have as much data as you wish you'd had to go on.

I want to turn it over to Stu Rubin.

MR. RUBIN: Thanks, John. My name again is Stuart Rubin. I'm a Senior Technical Advisor in the Office of Research and I'm also the PBMR Project Manager. First meeting with Exelon with on April 30 and our second meeting is scheduled for next week, so we're just starting our review. Can I have the next slide, please.

This next slide summarizes the objectives for the pre-application review. First of all, the objective is to evaluate the information that we're going to be receiving from the applicant on their design and their proposed new technologies and their regulatory process and framework for planned licensing. From that review we will identify where the information and the proposals appear to meet our expectations and needs for licensing of PBMR but we also intend to identify where there are gaps, gaps in the information on the design or design basis. gaps in the technology basis or the demonstration of that technology or the plans, therefore, and shortcoming that may have existed in their proposals for a licensing framework.

From those differences, we will endeavor to lay out the guidance and requirements that the staff and the Commission feel needs to be in place in terms of additional information and additional actions that will be needed to allow the design technology and framework to be acceptable as a basis for licensing.

The second objective is to develop an NRC core technology capability and capacity to conduct an actual licensing review. We are not doing a licensing review. We're doing kind of a feasibility licensing review. But should that feasibility prove positive and there is a decision to move forward, then the staff needs to be ready. So we will gain that capability from this work that we're now embarking on as well as additional training and the development of contractor capabilities, et cetera. Next slide, please.

This next slide identifies the significant review guidance and references that will be used to conduct the review. First of all, very important high level guidance and expectations for such a review and, for that matter, a licensing review are contained in the Commission's policy statement on advanced reactors as well as there is an additional NUREG document 1226 which provides additional staff implementing guidance for that Commission policy.

In general, the policy encourages innovative designs and innovative safety criteria but you still need to satisfactory consider such traditional aspects of our regulations, the application of the Commission's philosophy on defense and depth, safety goal policy, severe accident policy, application of industry codes and standards.

Also in the case of innovative designs, new technologies, demonstration testing, a prototype plan is particularly encouraged. Additionally, we will draw upon previous pre-application review experience as well as a safety evaluation report, a draft safety evaluation report, that was completed for a similar advanced HTGR design that was proposed by DOE in the mid 1980s. When one looks ta that design, one sees that the passive design features and safety characteristics of that plant are in many respects quite similar to the PBMR design and safety characteristics.

I would mention that kind of an underlying foundation for this entire effort will be an emphasis on traditional engineering and traditional design analysis viewpoints. The quality of design, conservatism of the design and analysis assumptions and safety modules. Again, our key objective is to identify the key issues that need to be addressed at the licensing stage.

Next slide, please. This next slide is intended to convey the broad scope that we have planned for the review. For example, in the fuels area we plan to carefully at the experience base and the analysis basis for the fuel design and to assess the fabrication processes and manufacturing plans for the production fuel. We also plan to look at the operating experience program and plan fuel performance demonstration and testing programs, not only on prototype fuel but that which would apply to fuel manufactured in a production facility as well as looking at plans for monitoring performance of the fuel in reactor.

Just to mention a couple of others in the nuclear design area, for example. Since the PBMR is designed to have passive shut-down characteristics, we intend to clearly assess how this will be demonstrated and, among other things in the nuclear area, we'll assess how well power distributions can be predicted for the PBMR -- moving fuel pebbles. In the thermal area, since the reactor there too is designed for passive, in this case, accident decay heat removal, we'll evaluate the effectiveness of these design features and, among other things, assess the capability to analyze temperature distributions during events as well as there are plans for verifying these tools including plans for using any prototype testing to benchmark the codes.

Just to mention a few others. The full scope testing plans that may be conducted we'll be looking at extremely carefully to look at what is to be included and what credits can be allowed by that testing. The planned PRA and there is an expectation that a PRA at some level will be provided for the plant. Certainly we'll need to get that kind of information in looking at any proposed framework for determining regulatory requirements.

Another important area will be the postulated events that will be applicable to the design. Certainly if one puts in or takes out certain events, it can affect the seriousness of the impact on fuel behavior. Next slide, please.

This next slide summarizes the overall process. My understanding is that we're not going to get an up front design package or, you might say, a preliminary safety analysis package from Exelon and so our plans are to kind of roll out the review on a month to month basis so a plan is to conduct monthly meetings with Exelon and the purpose of each meeting will be to allow the staff to get introduced to different topics through presentations from Exelon and subsequently to have that information provided formally on the docket and then to have the staff review that information and feed back its needs for additional information.

Again, we had our first meeting on the 30th at which Exelon discussed its plans for submitting formal proposals and basis for those proposals to mitigate or to eliminate certain requirements in the licensing process that they view as burdensome to a potential PBMR licensing. Those formal docketed proposals and bases have been submitted and staff is now reviewing those.

With regard to the proposed framework for determining regulatory requirements, that was discussed. We do have a description of that framework and the staff has developed its questions on that first proposal and fed that back to Exelon and we'll continue to dialogue at our next meeting which is next week. Again, future meetings. We're going to discuss traditional engineering design and design analysis areas such as nuclear thermal design. We plan to have meetings on fuel cycle safety and plant PRA, classification of SSCs and the like. Prototype testing is certainly going to be a major topic.

Again, we'll identify additional information after each of these kick-off meetings, you might say, that we'll have on a periodic basis and then that information will be documented and we will review that. So we will kind of continue our reviews and at some point, in addition to these public meetings, these meetings are intended to allow stakeholder comments at the end of each topical area so we can get some input from stakeholders on an ongoing basis. But in addition to that, we also plan to have a workshop that's specifically intended to invite in stakeholder comments on any and all areas.

We also clearly will be meeting with the ACRS and ACNW as we have completed our preliminary assessments to obtain advice and input and ideas that we need to consider before we go final and also as we progress through these reviews, we will inform the Commission in SECY papers of our findings and the staff positions and recommendations in various areas and then we'll feed back. Once we get Commission feedback sa guidance, we'll notify DOE and Exelon as to our positions and guidance in these various areas.

I would mention that as far as the Commission is concerned, in those areas where we view Commission policy decisions as necessary to establish licensing requirements such as in the containment design requirements or emergency planning requirements or a number of licensing process issues and legal and financial issues, the SECY paper will be a Commission policy decision paper. The staff will present its findings and recommendations and then we will obtain Commission decisions and guidance and then, following that, we'll be back to Exelon on the NRC's requirements in these areas.

The next slide, please. This next slide lists the technical resources and regulatory expertise that the review will utilize. Our strategy basically is to draw upon the best expertise that's available within the agency in both power reactor licensing and applicable HTGR design and technology expertise and to supplement it where possible, where resources allow, with additional outside expertise and experience. In each area, we intend to form a group of one to several part-time staff who will review that area and, if possible, to supplement it with contractor support.

For example, in the assessment of Exelon's risk-informed framework for making licensing decisions or establishing licensing requirements, we formed a review group of research staff and NRR staff as well as OGC staff and we do have contractor support identified familiar with risk-informing processes here in the agency.

I should point out that some members of the staff who will be working on this review also participated in the previous pre-application review of the DOE-sponsored modular HTGR in the late '80s. We also have the benefit of a rather complete draft safety evaluation on that review and that provides good resources as to the issues that one would want to take a look at and kind of a template for going through this review.

The design and operating experience of Fort St. Vrain will also be factored into the review, and we also plan to meet with NRC's foreign partners with HTGR design and operating experience, especially those with expertise and experience in coated fuel particle design and fabrication, radiation and testing experience and those who have design and possibly operating experience with the passive design features and safety characteristics.

Finally, in addition to Exelon input, we'll endeavor to get stakeholder input from federal workshop and to get ACRS and ACM input. Next slide.

This next slide lists some of the design and technology in regulatory areas where we expect there to be significant challenges in developing the guidance and the requirements for licensing of PBMR. A significant area will be the development of the guidance on information and actions for adequately demonstrating acceptable fuel performance and fuel integrity and demonstrating fission product retention capabilities over the life of the fuel and over the life of the plant and over severe event conditions.

One of the key points in all of that, as I mentioned, will be consideration of what are the design basis events and, beyond design basis events, that the fuel will need to be analyzed. Another area, just to mention one, is the guidance and requirements that the staff will look to develop for assuring acceptable performance of the core graphite components and reactor system pressure boundary metal components at the operating temperatures and levels of neutron flows are expected over the life of the plan. Again, the effectiveness of the design features, the passive design features, what kind of guidance we will need for adequately demonstrating. That will be another area that we'll be looking at.

Among the Commission policy issues, and I've tried to identify those with asterisks, the needs we believe will require a Commission policy decision are, for example, the possible use of a mechanistic approach to the source term. What are the postulated design basis events and, beyond design basis events, we need to postulate. The need for a leak tight containment. Whether that's what will be required or whether a confinement type structure with controlled and filtered release would be acceptable. That's clearly going to be a Commission policy decision.

And again, this question of using risk information to determine licensing requirements. That is new and we feel that that ultimately will require a Commission policy decision. Next slide, please.

I'd like to review our scheduling plans for the PBMR review. I would like to mention there are a couple of corrections on this slide. First, the third bullet should read "feedback on selected processing issues" and the fourth bullet should read "feedback on regulatory framework, financial issues and remaining licensing process issues."

As I mentioned, we kicked off the review on the 30th and we plan to complete the entire review in 18 months which would put it out to around October of next year. We're going to have monthly meetings with Exelon. We intend to get written follow-up documentation on what's presented and we plan to periodically feedback, as I mentioned, to Exelon our policy and positions on these topics. Again, we also plan to meet with the ACRS before we do all that.

So in just going through these feedback milestones, by this August or September time frame, we will endeavor to provide Exelon, to the extent we can, the staff's guidance and it's positions on the licensing process questions involving the early site permit proposal, combined license and design certification for initial PBMR facilities. Also by the end of this year, we will endeavor to provide Commission policy decisions and guidance on the proposed risk informed approach for making licensing decisions and the legal and the financial issues and the balance of the licensing process issues.

Within 12 months, we expect to feedback non-Commission policy level positions involving the technical and the regulatory and technology areas and then finally by the fall of next year, we will intend to provide the results of the Commission policy decisions on these major design and technology issues to the containment design requirements, emergency planning, source term, et cetera. Next slide, please.

This is kind of a repeat of what John talked about. Again, an objective and a by-product, if you will, of this review is to develop the infrastructure to effectively and efficiently conduct an actual licensing review on a PBMR. These kinds of development activities are fundamental to the role of research in supporting the agency's review of advanced reactor licensing. And so we plan to develop a training course with the support of contractor in HTGR technology. Our first class is hopefully going to take place this fall. We will be developing analytical tools for the analysis of designs such as the PBMR.

Also, hopefully going to have as an outcome a regulatory framework for conducting a licensing review of PBMR and possibly one that involves a risk-informed approach for making licensing decisions. And the other thing is we will identify where we might need independent testing and experiments on things such as the fuel performance and possibly the need for additional industry codes and standards for designs such as the PBMR. That's all. Thank you.

DR. KRESS: Thank you. Any questions?

DR. GARRICK: This is probably the question that I was half asleep on when George asked the question about the risk assessment. But you mentioned that on the PBMR you're going to get a risk assessment. What's the nature of that? Has that been requested?

MR. RUBIN: We have urged Exelon to provide as much information on the current risk assessment that they've done for the plan to support our review of this risk-informed framework for making licensing decisions. I wouldn't call it a risk-informed regulations framework as the extent of wholly replacing Part 50 but we think we now understand that this framework is not quite going to do that but will through risk insights be able to identify systems requirements for mitigation, prevention, the level of redundancy in those systems, which systems should be designated as safety significant and also things like what are the special treatment requirements on the system. But we're not talking about a regulations framework which covers all of Part 50.

But to answer your question, we have asked for that and we've also asked, to the extent possible, that we get information on the design itself. We have not yet, except for these kinds of viewgraphs that we've seen today, gotten what I would call a significant design description and principles of operation document from Exelon. I think the staff would very much like to get both a PRA and a design description so we have a context for reviewing this framework. It is on our schedule. We talked about that. It's not now but it is later.

DR. GARRICK: The thought is that it seems to me there's a possibility of a very much missed opportunity here. If you're talking about gearing up to license for advanced reactors, I can't imagine, given the history of pushing for performance-based, risk-informed approach here, of not being further along than you apparently are in establishing an infrastructure for doing that and, if there was ever an opportunity and a place to start it, it would be with the advanced reactors. I'm kind of shocked at the words I'm hearing. Possibly, maybe, a list of 500 other items here, 400 of them would be in a good PRA. I'm just kind of struck by this passiveness that comes across, to me at least, with respect to getting serious about practicing what you're preaching.

MR. FLACK: I agree with you. The PRA is an important piece that we still need to get. A lot of the underlining structure of that PRA is going to be in a sense driven by the success criteria, as you know, and the cost of fuels in this context is going to be extremely important. So you're absolutely right. We're ultimately going to have to put all this in perspective, and we're sort of going into it step by step. We had pushed the fuels issue up though because a lot of -- you know, understanding that is going to play out in PRA.

So I'm not too concerned that we don't have it right at this moment because in a sense it's going to take a while before I think they come up with a good one. I mean they probably will give us one, but I don't know how good it will be if we ask for it right now anyway. So I don't think it's holding us up any.

DR. GARRICK: Well, I made my point.

DR. WALLIS: Can I try to make a similar point? I listened to NRR and RES. Both parts of the agency are looking at what capabilities they need to develop to respond to a new design like GMR. So there's a tooling up. There's assembling expertise, there's building up infrastructure and all kinds of details. Seems to me that you're always going to be playing a long game of catch up with industry unless you have some other framework which is inherently more adaptable to any new technology and it seems to me that this framework has to be more based on risk information. It has to have a structure which puts risk in the forefront. Otherwise, you're going to be going through and building up a tremendous amount of deterministic type stuff which is then particular to every design, and it's going to take too long.

MR. RUBIN: Yes. I would absolutely agree that the time is now right to move forward quickly, as quickly as we can to develop this kind of a framework. Eighteen months ago, if someone were to propose what we're talking about now, you'd get a yawn from them because we did not know that there were such an interest that was going to be around the corner. But now that it's here, we agree that it's --

MR. THADANI: Stu, if you don't mind, pardon me for interrupting you. But I think we need to recognize that Part 52 for design certification requires the applicant conduct a probablistic risk assessment to provide that information to the agency to learn what the insights are to utilize those insights in the design. The only difference would be that under Part 52 it does, as Jerry Wilson said earlier, it does take you back to Part 50, Part 20 and so on. Now what we're talking about is an opportunity to really start with a clean sheet of paper and to build in risk insights up front. But anyone coming in under Part 52 design certification would be required by regulations to conduct a PRA. There are a whole host of other issues. Maybe we'll get into these issues later on during panel discussion. But I think there should be no misunderstanding what the Commission's expectations are.

DR. APOSTOLAKIS: But the PRA the way things are now could probably be one input to an integrated decision making process, would it not?

MR. THADANI: Again, it depends on what level of design information you have and the quality and robustness of the PRA. You could establish, it seems to me, a conceptual approach which would use probablistic thinking and then you could get into some design specific considerations driven by the level of information available. How far you can satisfy some conceptual set of requirements. We're not there.

One of the points I wanted to also say was we need to understand that while we talk about this small group that John Flack mentioned, we're just getting started and we're very sensitive to make sure before we go too far, we have Commission approval before we expend any significant resources. So all you're hearing is reporting to you on some of the meetings that have taken place and not really intensive thinking that is necessary. We will go through that process once the Commission does approve what John was talking about under SECY-0070.

So all these questions and issues you're raising I believe will be part of the process that we'll go through. The most significant being I think most of us are in agreement with what's being said. We want to try and maximize risk-informed thinking up front, clean sheet of paper kind of approach, rather than be overly influenced by existing structure.

DR. APOSTOLAKIS: Maybe we're getting into the panel debate here but I must say that I second Dana's observation earlier that we've heard very little about PRA today, and I'm under the impression that there is a gap between the staff's thinking and the industry's thinking. I mean most of the industry people who made presentations said, and we will do a PRA, whereas here we are saying we want the risk-informed and performance-based system and so on, so I'm not sure that the industry and DOE appreciate how important risk-oriented thinking is in both the design and licensing of these reactors.

I'm sure they will say no, they do realize it, they do know and so on, but it didn't come across from the presentations. I'm talking about quantitative risk assessment. Don't tell me that we're thinking about safety and we're designing against that.

MR. PARME: No, absolutely not. I want to make it clear. You were out of the room at the time, but we made it very, very clear that our intent on GT-MHR is to pick up where we left off in the mid '80s and I spent some time going through exactly that using risk assessment techniques and a risk assessment to build up our safety case. We believe that had to be done for a new reactor type and was the direction we planned on going. I understand you're busy and may have been out, but I want to make it clear that industry agrees with you completely.

DR. APOSTOLAKIS: I'm happy to be corrected. Thank you.

DR. KRESS: It sounds like we're almost in a panel discussion. I'd like to take a five minute break before we do the actual panel discussion to give us time to do some musical chairs and reorient. So five minutes.

(Off the record for a nine minute break at 6:16 p.m.)

DR. KRESS: Let's please come back to order. This is the time to ask questions and to make comments and get your points in. We don't have a particular protocol. I don't think we're going to have each member make preliminary comments. I'll just open it up for questions and let anybody who wants to.

MR. THADANI: Since we're talking about the PRA, it seems to me that the way we talk about PRA right now is being mentioned in a way that -- because first of all, it seems to me we are looking at these new designs with old criteria. They were talking about new PRA -- design and using some of the criteria here to get -- additional burden and I feel that unless we -- try to set a different kind of performance measures, for example -- we're going to simply -- requirements which may not be necessary.

DR. KRESS: Does anybody on the panel want to respond to that?

DR. BONACA: Certainly the Commission has been very clear, I think, in articulating its philosophy and moving more and more towards risk-informing regulations even for the operating reactors. So it's very clear that when we're going to these new advanced designs, you're exactly right that risk-informed thinking has to come in up front, recognizing some limitations. One has to be careful that one understands what the uncertainties might be. We have a tremendous opportunity now to start with that thinking up front such that it can then identify potential areas where we need additional information.

For these new technologies, I would expect we would put together a number of panels to look at phenomenon, see what the important phenomena are, identify those, rank then and rank them understanding what the risk implications might be. And it seems to me that would be a good way to define not only the kind of testing programs that would be appropriate but also to make sure that the tools, the analytical tools that we have are robust enough to give us that analysis capability which can then be turned around back again trying to understand what the risk implications are.

So I would expect we would go through that process. Clearly, it's a policy issue. You heard earlier about potential petition coming in from NEI. I don't think they are thinking petition option any more, but I'm not certain. But we are as part of our plan that we've been talking about that we've sent to the Commission, this is one of the issues and I would fully expect support. That's the way we would proceed.

DR. BONACA: The reason why, just to complete the thought process, my sense, from what I've seen and we're going to have maybe an SAR coming in with Chapter 15 with all the traditional analysis coming in. Okay. That's the understanding I got from the presentation.

MR. THADANI: I think we are open, up front to what I described as conceptual model pretty much will have to take into account more than the Commission's safety goals because the surrogates that we use from Commission safety goals have two points essentially: core damage frequency and large early release. Clearly, we need the whole spectrum which means you do have to have the whole sort of CCDF, the complimentary cumulative destruction function. If you start out that way, the questions that we would then face would be is that the level at which you can say that's technology neutral safety -- so to speak. And then if you were to go design specific considerations, is that when you come up with general design criteria or something else?

It is at that point that that information, seems to me, ought to help us come to grips with what are the design basis events. They need to be driven by this safety philosophy that has to be let out up front and which, in my view, is more than what the current safety goal policy statement says.

MR. PARME: Let me add, in response to your question, whether it's a burden. Going back to the DOE submittal of the 1980s. The PRA that we used at that time was not a significant addition to our task. In fact, it was the forerunning analysis. The PSID, preliminary safety information document, which accompanied the PRA and had deterministic analysis, was pulled out of the PRA. The PRA gave us the uncertainties and the understanding of this up front. Obviously, two documents cost more than one but, in fact, having started -- and in fact, I can recall in 1982 working with the Germans, having evolved our PRA with our design and the first cut being I think it was a 25 page memo and having evolved that through the early '80s as we had the design, it was not a large incremental cost on the thing.

The only thing that became a burden was having gone to the Commission and having a rationale for why we did all these things and then to have the Commission come back. It was a good interaction but when the Commission came back at times and you got a response, we don't agree, and the reasons were often there was no point to discuss why they didn't agree with what we had done. That was frustrating. That was a burden and that cost more money than doing the PRA.

DR. POWERS: Ashok, you bring up phenomenology and I'm delighted that you did because I don't think it's possible to do technology independent regulation. Sooner or later you have to get down to how the system really works. I think that's going to raise a real headache for the NRC because you don't have the wealth of phenomenological information about these new designs that you have for your existing designs. Seems to me that indeed frequency consequence curves look like an appropriate approach to go. That means you have to go to something like a level 2 type analyses and you're going to have to make a decision along that way at which point you have to do your own confirmatory experimentation, your own confirmatory codes.

It looks to me like in the past we've done that on a catch as catch can basis, but if there are indeed going to be these multiple kinds of designs coming to you for at least consideration of licensing if not actual certification kinds of applications, we'd better start putting in some sort of a process by which we can make these confirmatory experimentation and analysis decisions in predictable kinds of fashions. That just seems like a priority that the ACRS and your organization needs to start kicking around outside of the more formal structures because it's going to be necessary in spades. You're going to have lots and lots of head knocking taking place where licensees presenting test results that say, gee, I present you these results because I have assumed that coated particles failure only depends on temperature. And that's a fine assumption to make but you're going to want validation of that.

The question is do you get that validation or does the licensee get validation? It's a question that's going to have to be answered some place.

MR. THADANI: I agree. First of all, I think it's very clear -- and I brought this report just to really make a point I think fits in nicely with what you said. This is work we did on AP600 in cooperation with Jerry in Japan. It was at ROSA facility and I can tell you it was extensive involvement. I think we did 20 separate experiments. Some of the work that was done here led to actually changes in design and impacted schedule in a positive way because we were able to use this information to respond to many of the ACRS questions, as a matter of fact.

My own opinion on NRC's need to do independent testing comes from the fundamental view that you get deep understanding by doing things, not just by reviewing other people's work. That's a fundamental point. Second, there are some areas in the fringes which are not necessarily required by regulations requirements. I personally think it's appropriate for a public health and safety agency to sort of poke and probe at the fringes. Try to understand where the thresholds might be. That would be independent testing.

In terms of confirmatory work, it's clear to me that there are some very crucial areas. Fuel or fuel cladding may be very crucial from the metal things to safety. It's the most important barrier we're talking about. I think it's appropriate for the agency to do some independent confirmatory testing, even if the industry were doing some testing in that area. It's amazing sometimes how much you learn by conducting such testing. How certain issues come to surface that really get you to go into a fairly challenging dialogue sometimes as to how one would proceed.

Analytical tools. Historically we have really gained a great deal by our ability to do independent analyses. And so I personally again am very much in support of making sure we have those analytical tools that we can employ and when we get results, try to see if there are differences and sort of hone in on what they key issues might be.

So basically I do agree with you but that's why I think PIRTS are going to be very important for us to know where should we focus really our attention in this area?

DR. POWERS: I think the program you've carried out in high burn up fuel has shown you that the PIRT technology has applications for getting your staff up to speed beyond the thermal hydraulic area. At some point we're going to have to come down to pretty hard and fast decisions on where to investigate. I think you're right. Fuel is going to be a head ache here because we just lack the kinds of experience with this kind of fuel that we're going to have to have to feel comfortable.

DR. KRESS: I partially think the time frames are such that to get the kind of data you want on particularly these coated particle fuels, that is a difficult task because we're talking about a fuel that's radiated to some burn up level and get appropriate statistics for 15,000 per thing, it has to be put in a reactor, it has to be run through the temperature transient that you're dealing with and you're looking for two things. You're looking for fuel quality in the first place and then you're looking for what do the transients do to the fission product release and what sort of model can you put on that fission product release to get a source term out of it?

I just don't think we have the time to do confirmatory research in that area. So I think NRC is going to have to decide on how they're going to deal with those particular issues. I think they'll have to rely in this case on existing data and existing fission product release models and existing analytical tools.

DR. POWERS: Stun me if you could, Tom. I mean we've got basically models based on chemical diffusion and poor diffusion in a situation where thermal diffusion is going to be dominant.

DR. KRESS: Exactly.

DR. POWERS: I just don't think you can. I think you're going to have to do tests and it's the classic story of --

DR. KRESS: I'm not even sure we have the reactors to radiate these things.

DR. POWERS: It's the classic story of planting trees. The best time to plant a tree is 20 years ago. The second best time is right now.

MR. SPROAT: Let me just say in this whole area of particle fuel testing, there's no doubt in my mind that the application of particle fuel and pebble bed application if we go forward here in the U.S. clearly will have to have a well-documented fuel testing qualification program that answers some of these questions. However, there is significant data, both operational data and test data, that exists on particle fuel including naval reactors, and I would severely question the need to go back and replicate and duplicate at great expense and great delay all of that information. I think it's incumbent on both us as the applicant and I think it's incumbent on the regulator to be able to go back, extract the relevant data out of the existing vast bodies of data, determine where the gaps are and focus the additional testing on those gaps and not reinvent the wheel.

DR. KRESS: Is the naval reactor --

MR. SPROAT: To some extent, yes. Absolutely.

DR. KRESS: -- How do you see the role of a prototype test in this respect in terms of validating the codes and the assumptions that go into it?

MR. SPROAT: As we took a look at trying to license the PBMR here in the U.S. Clearly, I think I said in my presentation, we can't go for certification first in this country. We have to go for a COL first. We fully expect that as we go through the licensing review process here with the NRC, there will be a number of technical issues that will be unresolved or open as we go through the review process which will need to be resolved during the start-up test program of the demonstration plant in South Africa.

It's one of the great advantages we have with the program, at least as it's currently envisioned, which is with the demonstration plant in South Africa leading whatever we do here in the U.S. We'll be able to utilize that demonstration reactor to reduce significantly a number of the uncertainties associated with the codes, with the codes, the fuel performance, that type of thing.

So what we would like to do ideally is to get far enough through the review process with the staff here so that the key unresolved issues are identified and then we can jointly figure out with the staff and with the South African project how the South African start-up test program needs to be modified with the appropriate acceptance criteria so that the appropriate testing is done during that one year start-up test program that's in the schedule for the South African reactor and put those issues to bed before the license is issued for here. We think that's a reasonable approach.

DR. GARRICK: Has this data that you refer to been documented and peer reviewed, et cetera?

MR. SPROAT: I'm not a fuel expert, and I personally have not reviewed the fuel data. But the Germans spent over several billion Marks on particle fuel testing and the ABR. They had their experience in the THTR. Obviously, in the U.K. gas reactor program, particle fuel was also tested there and utilized, and we have the naval reactor programs here in the U.S. and over in the U.K.

In addition particle fuel is currently being fabricated in China, Japan, Russia. I mean there is a significant amount of international data on this fuel. Now, does it all necessarily envelope the exact operating conditions of the PBMR as we're designing it? Personally, I'm not sure and clearly, if we were to go forward with the licensing process, we do need to make sure that it's appropriately enveloped, see where the gaps are and design the testing qualification programs to cover that. But I think we'd be amiss if we walked out of here today and left the subcommittee with an impression that this particle fuel stuff is all new and there's not a lot of information about it because that's not the case.

DR. FORD: I'd love to hear the opinion of the panel about the whole question of materials degradation, time-dependent degradation, especially with a risk-informed regulatory environment we're going into. I heard no one talk at all about it. Every one of the designs that we've been talking about in other countries, Southern Korea to the advanced gas reactors in Britain and light water reactors in this country, of course, have all undergone cracking or embrittlement problems of some type or other. You mentioned the -- chrome situation. For the IRIS, I didn't see anything at all in that design to say that you would minimize the frequency of cracking events. You may influence the impact them but not the frequency. Could someone address this?

MR. SPROAT: Let me start off and just

talk about the PBMR materials. Clearly, one of the areas we've looked at very closely in our involvement in the project is materials because you're looking at core outlet temperatures of 900 degrees Centigrade. The ABR in Germany ran the bulk of its career at 950 degrees C. core outlet temperature. If you're familiar with gas reactor technology at all, clearly, you know that graphite aging under irradiation and temperature is a an issue and how graphite reacts under long-term irradiation where it first shrinks and then re-expands is a phenomenon that's known but it's very much specific graphite material dependent.

So my answer to your concern is, #1, that it's absolutely a valid concern. #2, that it needs to be addressed in detail during the detail design and it needs to be addressed via the appropriate materials testing qualification program during the design phase and the development phase of the particular technology that you're talking about. We've been working with the South Africans to try and make sure that their thoughts about what needs to be done in their materials testing development program coincides with ours, based on what we know are issues we'll have to look at. As part of our application if and when we come in, we would have a materials test and development program in there.

Right now, just to give you an idea, graphite is clearly one area. Some sort of carbon carbon composite insulation material that we use in the hot duct piping is clearly another area. Fuel we've already talked about. The material we'll use in the high pressure compressor blading for the turbo compressors is another. But again, we're in that preliminary design stage where those issues and the limiting conditions for each of those key materials is just now being identified, developed and a mitigation strategy put together for them.

MR. PARME: Let me add to that. Forty five minutes is kind of tough to cover all the subjects when you describe a design, but if you pull up the plan view of the prismatic block core, you'll see that both replaceable and permanent reflector elements are noted in there from the experience through the '70s and '80s and radiation experience with graphite type of age and radiation and who's changed the block is known, and that's designed for.

Right now in our program in Russia, one of the primary things it's looking at is overhaul of the turbines. We're well aware the turbines will not last the life of the plant. In fact, nowhere near that.

And it's designed to come out. It's designed to be serviced and currently we're looking at various alloys, alloy possibilities for the blade but also the possibility of whether we should go to turbo machinery replacement or is it possible -- mind you, these turbines, there's some plate out of activity on them, especially the turbine itself -- whether we can go in there though and change the blading out. So there are a number of these things being looked at but, as I say, I wasn't the materials expert. They sent me, the systems engineer and safety. They said that's what they'll want to hear about. But these things are being looked at as the design proceeds and certainly I think the industry experience says you need to look at that up front.

MR. CARELLI: You asked about the IRIS. Again, IRIS is the youngest design here and, very honestly, I didn't look at the materials because right now this is not a top priority. In the case of the light water reactor, we rely on what it is the body of the light water reactor. There are two things with IRIS -- light water reactor and the first one is our power rating is much lower. We are talking probably half of the power rating of LWR. Actually, we'll do even in AP600. So a neutral environment is more benign.

The other thing is what I showed you earlier, the capability of putting internal shields. For example, the vessel. We don't want to put numbers but the vessel in IRIS should last a lot longer than the vessel we have in the present LWRs because basically there is no radiation in the vessel. So there is no question that the materials is an issue and, in the case of IRIS, will be especially an issue on what is new. Like the steam generators, the pumps that are going inside the reactor. Those are the ones we'll be focusing on. We already started already looking for the steam generators. In the case of the pump, I mentioned the spool pump we have.

The only reason we've been holding on putting that as a reference design is because of materials issue of the bearings at high temperature. So definitely we're going to look into that. Again, it is the kind of thing that we can not look at other materials once we have a design. Our first emphasis is to have a design. Now we have a design and we're going to look at the materials.

One thing we've done, for example, for the extended life time core, the one that reloads, the cladding most probably is going to be a stainless steel. So we've been looking at those issues.

MR. THADANI: I just wanted to make sure. John Flack gave us some idea of the issues. High temperature material issues are amongst the top issues, particularly when we are talking about getting temperatures of 900 C. to 1,000 C. Not only degradation, aging would be an issue, but we're also going to be looking for some other kinds of challenges such as thermal shock external to the vessel, for example. What are the potential impacts of things of that sort when you have material at such high temperatures? So it's going to get a fair amount of attention from us as well.

DR. FORD: I guess as a follow-up question, Doctor Thadani, you weren't here when I asked the question this morning. That's all very well and good, but you've got a severe weight limiting step with the number of people who can do this job adequately in the time that you have. I think you've got a major problem. We all have a major problem in that particular area.

MR. THADANI: It's a challenging task, I agree.

MR. RAE: Let me add my two bits to it. The devil's in the details. At least we at G.E.

believe that materials are a big issue and we have tried to keep the design within the range of all the experience base that we have right now. We have a second line of approach which is to make sure that the internals are removable, so we are making the internal designs such that they are easily removable in case whatever you taught us we didn't learn properly.

Finally, on the sodium reactor. Unfortunately, I can't answer that question. That's a little further out in time.

DR. KRESS: I hope I made it clear that people in the audience are welcome to enter into this debate also if they want to make a burning comment or question.

I have a question for you, Ashok. You mentioned one possibility for frequency consequence curves could cover most of the regulatory objectives and I'm confident you can derive the end points for those using the safety goals. I'm not sure you can get slopes, but you can get the end points.

The question I have is in view of the advance reactor policy statement which has an expectation, I think, of a better level of safety, what safety goals are we talking about? Are we talking about the ones in the utility requirements document or the ones we have now that we use in 1.174?

MR. THADANI: Remember, 1.174 is only looking at deltas.

DR. KRESS: No, it looks at -- also. But it's debatable.

MR. THADANI: Yes. I go too far. But I think I learned from experience, as we all do. When the EPI requirements document was submitted to NRC, it had some objectives for designers. One of the objectives in that was that the core damage frequency shall be equal to or less than 10^{-5} per reactor year of mean value. Let me be clear. And so on. At least at that time, the guidance we got from the Commission was very clear that it was driven by the statements in advanced reactor policy statement.

The view was the Commission expects these new designs to be safer. Expects these new designs to be safer. But that doesn't mean that we should establish requirements that make them safer. Their view was that we should not go beyond what the Commission safety goal policy statement says. That's the only background I have to go on at this stage.

Now we're embarking on some really quite significantly different arena. At that time, the Commission's decision, I'm sure, was driven by understanding what the margins were and what the various levels of defense that were provided. I think we will have to go back to the Commission. We'll have to go to Commission regardless. It's very clear to me that the one end point of the safety goals is not enough to develop risk-informed -- that's just not enough.

So we'll have to go back to the Commission and seek their guidance on how much farther we can go. At this stage, I can only tell you what we've been told up to now.

DR. KRESS: In that same respect, take, for example, the modular pebble bed reactor. They, I'm sure, show they can meet something like the early fatality safety goal with lots of margin. The question I have there though is -- and they could probably meet some sort of frequency consequence curve that you might establish to cover the full regulatory set of objectives. The question I have is how in that arena, how would you deal with defense in depth? Where does defense in depth come into play when you're asking someone to just meet a frequency consequence curve?

MR. THADANI: That's why I said that you can establish in a conceptual sense that you can't really answer these questions you're raising about defense in depth until you get to a specific design and until you understand where the uncertainties are to make some decisions.

DR. KRESS: You would relate it to the uncertainties in the --

MR. THADANI: It seems to me that's the most logical.

DR. KRESS: I certainly --

DR. APOSTOLAKIS: In this respect, would it be crazy to look at past history and say, boy, we were surprised four times in the last 20 years and we're going to be surprised again. The prudent thing to do is to really require defense in depth in which case, of course, extra measures of defense in depth, in which case you reduce the significance of the PRA. I wonder whether that's just an academic exercise or it's something real? The reactor safety study under-estimated significantly the importance of external events and design end point study show that these were very important. We were not paying much attention to the human element until Three Mile Island.

So this feeling that we are dealing with a new design, new concepts, we're doing the best we can with the PRA, we'll use it to the maximum extent we can. There's always this uncertainty about things, metaphysical things that we don't know about. Would it be prudent to add an extra layer there at the risk of making the design uneconomical? I think that would be a major issue, a major challenge, and I really don't know how to handle it.

DR. GARRICK: But, George, you do agree, do you not, that one way to address defense in depth is in the way in which you express your confidence about the parameters?

DR. APOSTOLAKIS: I do agree with that. What I'm saying is that my confidence may not be what the analysis shows. For light water reactors, it really took us what? a good 20 years to reach a mature representation in terms of risk matrix and so on. I don't think that anyone expects that tomorrow there will be a risk assessment for an LWR some place that will come up with something fundamentally different the way Indian Pain and Zion did or other studies later. It's mature now. We have reviewed it 1,000,000 times. We understand it. We have a significant experience and so on.

When you start with new concepts, I wonder whether that kind of thinking should play a role. I think that was the thinking in fact behind defense in depth to begin with, that we could not quantify. I guess I'm talking about something that you don't like, John. Unquantified uncertainty.

DR. GARRICK: You're right, I don't like it.

DR. APOSTOLAKIS: I know you don't like that, but it's a fact that this thing is there.

MR. PARME: Let me suggest there is one way of possibly -- I don't claim to have an answer. It's a difficult question to answer, but one of the things that we were thinking about. If you look at the '80 submittal it basically says below 5 X 10^{-7}. There's nothing else bounding us. There was no reason to analyze things below there except to sum up risk. But one of our thoughts -- we had the same question. Finish with conceptual design. You know there's a lot of uncertainty in the work you did and it's new design, too.

But I think one of the things that built our confidence was we just took them all to the worse case and made some simple assumptions at the bottom and what we did then with the risk assessment though is we could see what were, in a sense, not so much from a frequency point of view but phenomenologically. How bad could things go on us? We had that on the table on paper. We had the calculations that showed us. Once we understood that, we suddenly were not quite as worried, have we missed a frequency here by some amount? Have we misunderstood this? If the worse case reactivity accidents were only so bad and took three days before you really heated things up or if pumping steam from the other nearby reactors for several days into a scrammed reactor. I mean it's absurd but we could see what happened. And it sort of gave some feeling for what were the chances that we have missed something important?

Of course, our argument to the NRC was that's in the PRA. It's not frequency of concern. We don't want to be judged against this. But my hope was they could read the same document, too, and determine how comfortable they were or were not with the uncertainties that are bound to exist. As I say, I don't think it's a complete answer but it was one of the ways we tried to address it and I think it has merit. Just understanding what's sitting there --

DR. APOSTOLAKIS: I agree. I agree. I mean if that argument can eliminate all this uncertainty that I'm talking about, then great.

DR. KRESS: That, in essence, is a kind of uncertainty.

MR. PARME: It is. Yes.

DR. POWERS: I think that's something that we do too little in this field is to go look and see how bad things become if everything goes wrong. I will remind people that a lot of defense in depth comes about by asking the question, what if you're wrong?

DR. APOSTOLAKIS: On the other hand, you can't really push that argument too far because you end up with traditional deterministic --

DR. POWERS: You and I have written a paper in which we said don't push it too far.

DR. APOSTOLAKIS: Okay. Good.

DR. POWERS: Push it to the first level and stop, as I recall.

DR. BONACA: I was curious about this. This morning we heard a presentation from Doctor Slabber in which you were mentioning, for example, on fuel integrity, you are designing for anticipated transients, 10 X 2^{-2} and then to the range of 10^{-2} X 2^{-6} for licensing basis events and beyond that is analogous. Are you using PRA behind this analyses in licensing efforts?

DR. SLABBER: Yes. To answer your question, we are using generic values at the moment to get into the ranges. What we do and then deterministically we calculate the consequence and, in general, it doesn't take you out of the range which is prescribed by the licensing authority. So even if you've got some error bands which are quite large, it still, with this type of reactor, it keeps you way on the low consequence level so it doesn't really impact. But the question is yes, we're using generic--

DR. BONACA: And so you can use that PRA as a basis for justifying your analysis that you submit into the licensing area?

DR. SLABBER: Yes.

DR. KRESS: Ted, did you have a comment you wanted to make? You've been standing there a while.

MR. QUINN: Okay. I have a question. It's Ted Quinn. It has to do with process. To set the stage, a number of the vendors, the applicants today, have discussed the importance of the pre-application process. I'd just like to ask the ACRS or panelists. The going forward part of the next year or two as we look at it, in the pre-application process Stu Rubin put up a list of items that are very important, for example, to the PBMR. Any one of the applicants could have that similar list. As you go forward, they've also stated that the results of the pre-application review are very critical to their management or the process of going forward after this is done because some of the key issues that are being presented, some of which are technical and some are policy, can get decided as part of this process.

Is it clear to you, the ACRS, that sufficient information can be developed as part of pre-application that the staff can review it, that the ACRS can weigh in and that the Commission can approve policy issues such as EPZ and definition of some of the key issues as part of this so that the companies can go back and go forward with a detailed design?

DR. KRESS: Anybody want to take that one? I'll give my opinion. I've seen preliminary designs for most of these reactors. I've seen safety analyses for most of them and looked at some of the competitional tools that they've had. I think the answer is yes, that you can. I don't know. That's just a personal opinion.

DR. GARRICK: I think that there's a model for this with respect to Yucca Mountain. Why do you laugh?

DR. POWERS: Doesn't sound like a promising model.

DR. GARRICK: But a model from a process standpoint. Your question was a process question, and the question that is being tackled now with respect to licensing Yucca Mountain, is there a sufficient basis for there to be an application for a license? So that's an inherent part of the process, to establish that there is a basis for going forward with the license application. And it's a very systematic, deliberate and detailed process.

MR. THADANI: If I may. Certainly we think we can do it in 18 months. I just want to be sure that there's clear understanding of what it is that we will deliver. It's sort of what I would call some key technical issues or key policy issues. It would a roadmap basically to lay out what will it take, the kind of information, data, the need for tools and so on, what will it take for us to resolve throe issues? It's not that we have developed all the information and resolved, clearly not. It's just that laying out a roadmap as to what is it that we need so there's a clear understanding like the PBMR, there's a clear understanding of what the expectations are and for Exelon then to make some decisions.

So I think it's a good process. It really is. It not only helps Exelon. I think it helps us.

It helps our reviewers as well. Anyway, so I think it's doable.

DR. KRESS: I think we're getting tired and hungry. So I think at this point, unless someone wants to make a final comment, I'll recess this meeting until tomorrow morning. We start again tomorrow in this same room I think at 8:30 instead of 9:00. So the same room tomorrow at 8:30. We stand recessed.

(The committee recessed at 7:13 p.m. to reconvene tomorrow at 8:30 a.m.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: ACRS Subcommittee on Advanced Reactors

Docket Number: (Not Applicable)

Location: Rockville, Maryland

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

John Mongoven

John Mongoven
Official Reporter
Neal R. Gross & Co., Inc.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE ON ADVANCED REACTORS
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JUNE 4 - 5, 2001

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REGULATORY CHALLENGES FOR FUTURE NUCLEAR POWER PLANTS

	TOPIC	PRESENTER	TIME
1.	Introduction	Tom Kress George Apostolakis	9:00- 9:15 am
2.	Keynote Address	Commissioner Diaz	9:15- 10:00 am
		BREAK	10:00- 10:15 am
3.	DOE Presentations		
•	Overview and Introduction to Generation IV Initiative	W. Magwood (DOE)	10:15- 10:40 am
•	Generation IV Goals and Roadmap Effort	R. Versluis (DOE)	10:40- 11:00 am
•	Near-Term Deployment Efforts	T. Miller (DOE)	11:00- 11:25 am
•	Generation IV Concepts	R. Versluis (DOE)	11:25- 11:40 am
•	Next Steps Generation III+/IV	S. Johnson (DOE)	11:40- 12.00 pm
		LUNCH	12:00- 1:00 pm
4.	Generation IV Design Concepts		
•	Pebble Bed Modular Reactor	W. Sproat (Exelon)	1:00- 1:45 pm
•	International Reactor Innovative and Secure	M. Carelli (Westinghouse)	1:45- 2:30 pm
•	General Atomic- Gas Turbine Modular Helium Reactor	L. Parme (General Atomics)	2:30- 3:15 pm

	TOPIC	PRESENTER	TIME
		BREAK	3:15- 3:30 pm
•	General Electric Advanced Liquid MetalReactor and ESBWR designs	A. Rao (General Electric)	3:30- 4:15 pm
5.	NRC Presentations		
•	NRR Response to 2/13/2001 SRM on Evaluation of NRC Licensing Infrastructure	M. Gamberoni T. Kenyon E. Benner A. Rae A. Cubbage	4:15- 5:15 pm
•	Planned RES Activities	J. Flack S. Rubin	5:15- 6:00 pm
6.	Panel Discussion on Industry and NRC Licensing Infrastructure Needed for Generation IV Reactors	Panelists: J. Flack, NRC S. Johnson, DOE W. Sproat, Exelon M. Carelli, Westinghouse L, Parme, General Atomics A. Rao, General Electric	6:00- 7:00 pm
1.	Closing Remarks and Recess	T. Kress, ACRS G. Apostolakis, ACRS	7:00 pm

INTRODUCTORY STATEMENT BY THE CHAIRMAN OF THE
MEETING OF THE ACRS SUBCOMMITTEE ON ADVANCED REACTORS
WORKSHOP ON REGULATORY CHALLENGES FOR
FUTURE NUCLEAR POWER PLANTS
11545 ROCKVILLE PIKE, TWFN AUDITORIUM
ROCKVILLE, MARYLAND
JUNE 4-5, 2001

Introductory Remarks by ACRS Chairman, George Apostolakis:

It is with great pleasure and honor that I introduce our keynote speaker for this workshop, Commissioner Nils Diaz.

Before being sworn-in as a Commissioner of the U.S. Nuclear Regulatory Commission in August 1996, Dr. Diaz had 34 years in nuclear and radiological engineering, as a scientist, engineer, researcher, consultant, and entrepreneur. In the research and development arena, Commissioner Diaz worked from mundane light-water reactor safety and advanced designs to more complex space power and propulsion systems, and in the conceptual design and testing of futuristic reactors like the UF6, UF4, and U metal-fueled reactors for the Strategic Defense Initiative.

INTRODUCTORY STATEMENT BY THE CHAIRMAN OF THE MEETING OF THE ACRS SUBCOMMITTEE ON ADVANCED REACTORS WORKSHOP ON REGULATORY CHALLENGES FOR FUTURE NUCLEAR POWER PLANTS

11545 ROCKVILLE PIKE, TWFN AUDITORIUM
ROCKVILLE, MARYLAND
JUNE 4-5, 2001

The meeting will now come to order. This is the first day of the meeting of the ACRS Subcommittee on Advanced Reactors. I am Thomas Kress, Chairman of the Subcommittee.

Subcommittee Members in attendance are ACRS Chairman George Apostolakis, Mario Bonaca, Graham Leitch, Dana Powers, William Shack, Jack Sieber, Robert Uhrig, and Graham Wallis. Also, attending is ACNW Chairman John Garrick.

The purpose of this meeting is to discuss matters related to regulatory challenges for future nuclear power plants. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. Michael T. Markley is the Cognizant ACRS Staff Engineer for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the *Federal Register* on May 10, 2001. A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice. We have received no written comments or requests for time to make oral statements from members of the public regarding today's meeting.

In order effectively manage time and allow for maximum member, presenter, and public participation and sharing, the Subcommittee requests the following protocols be adhered to during the meeting:

- Presenters should be allowed to make their presentations without substantial interruption.

- Questions from the audience/stakeholders will be entertained at the end of presentation sessions, not the individual presentations.

- Members of the public/audience should use question cards provided. The ACRS staff facilitator, Mike Markley will collect comment cards, group like comments as practicable, read them into the record, and refer questions/comments to presenters and/or panel participants, as appropriate.

- It may not be possible to respond to all questions and comments. However, all questions/comments will be listed in the meeting proceedings following the workshop.

Opportunities for direct audience participation will be provided during panel discussion sessions each day. Microphones have been arranged for convenience of the audience during this meeting. It is requested that speakers first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

(Chairman's Comments-if any)

We will now proceed with the meeting and I call upon Dr. Apostolakis to introduce the keynote speaker.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Remarks of Commissioner Nils J. Diaz

United States Nuclear Regulatory Commission

ACRS Workshop on Advanced Reactors

June 4, 2001

It is a real pleasure to participate in this workshop to discuss regulatory challenges for advanced nuclear power plants. It is particularly appropriate that the Advisory Committee on Reactor Safeguards is hosting this meeting, at this time. The discussion on nuclear power has now fully entered the national debate on the future of America's energy supply, and nuclear safety is going to be a priority on everybody's agenda. The Commission relies on the ACRS for expert advice on the safety of reactors, existing or submitted for licensing. The recommendations of the Committee will be of particular value for the Commission deliberations on the licensing of new reactors. I will be presenting my individual views today. They do not necessarily represent the views of the U.S. Nuclear Regulatory Commission (NRC), except when indicated.

I want to premise my remarks with a few selected quotes from a "couple" of speeches during my tenure as a Commissioner.

- "There is no credible regulator without a credible industry. There is no credible industry without a credible regulator."
- "It is essential for the regulator to be cognizant of the technology. It is essential for the industry and technologists to be cognizant of the regulations."
- "Regulations need to result in a benefit or they will result is a loss."
- "My goal is to ensure the paths are clearly marked. A path that is clear of obstacles and unnecessary impediments, with well defined processes, will provide regulatory predictability, equity and fairness."
- "We are learning how to define adequate protection in more precise terms, and to define it in terms that make sense to the American people."
- "We have learned from our mistakes and we are bound not to repeat them."

At the 2001 US NRC Regulatory Information Conference, I said: "We might be asked, as would other government agencies and the private sector, to sharpen our skills, and improve our efficiency to meet the needs of the country". We have been asked. It is worthwhile to try to understand why the President and the Vice-President of the United States have brought nuclear power generation to center-stage in the debate on the energy policy for our country. Shown in Table 1 is a compilation of important aspects of the debate, summarizing what has changed in 20 plus years.

The NRC has been changing to meet the challenge of what must be changed and to strengthen what must be conserved. I submit to you that we have changed for the better, especially the last 3 years, and that improvements in regulatory effectiveness and efficiency are changing from goals into reality. But it has not been easy, and there are still lessons to be learned. I must say that there is one change that I believe speaks louder than words for the NRC staff and the agency as a whole: priority is now placed on what should be done better rather than on what was done wrong.

This is a cultural change that is needed to enable the consideration of newer, better and enduring ways to exercise the mandate entrusted to the NRC by the people of this country: to license and regulate the peaceful uses of nuclear energy, with adequate assurance of public health and safety. I believe that we are now capable of meeting the regulatory challenges that we face today regarding advanced nuclear power plants. The improved industry performance over the past decade has enabled the NRC to initiate and implement reforms that are progressively more safety-focused. Furthermore, it allowed the industry to concentrate resources on the issues important to safety which provided a sharper focus to regulatory improvements. Safety and overall performance, including productivity, became supporters of each other, with the clear and unmistakable proviso that safety is first.

For existing nuclear power plants, the list of profound regulatory changes and accomplishments, many done under the mantle of the so-called risk-informed regulation, would occupy the rest of this meeting. Five of them stand out: the revised rule on changes, tests, and experiments for nuclear power facilities (10 CFR § 50.59); the new risk-informed maintenance rule (10 CFR § 50.65 (a)(4)); the revised reactor oversight process; the new guidance on the use of PRA in risk-informed decision-making (Regulatory Guide 1.174); and the revised license renewal process (10 CFR Part 54). The list is growing. About two weeks ago, the Commission approved COMNJD-01-0001 instructing the staff to give high priority to power uprates and allocate appropriate resources to streamline the NRC power uprate review process to ensure that it is conducted in the most effective and efficient manner. All of these and most of the other regulatory improvements conform to the Commission's decision to focus attention on real safety. The resulting improvements in rules, regulations and processes, including changes to the hearing process and enhanced stakeholders participation, are assuring the nation that a fair, equitable, and safety-driven process is being used.

I mentioned risk-informed regulation as an important component of the changed NRC regulatory structure. I want to be sure you know what I mean when I use the term risk-informed regulation, so I am going to present you with my own, personal definition of it:

> Risk-informed regulation is an integral, increasingly quantitative approach to regulatory decision-making that incorporates deterministic, experiential and probabilistic components to focus on issues important to safety, which avoids unnecessary burden to society.

The definition can also be used for risk-informed operations, risk-informed maintenance, risk-informed engineering, risk-informed design....

For new license applications, much groundwork has been done, and a lot of it is useful to address today's issues. In the statements of consideration for 10 CFR Part 52, the Commission stated that the intent of the regulation was to achieve the **early** resolution of licensing issues and **enhance** the safety and reliability of nuclear power plants. The Commission sought nuclear power plant standardization and the enhanced safety and licensing reform which standardization could make possible. In addition, the 10 CFR Part 52 process provides for the early resolution of safety and environmental issues in licensing proceedings. The statement of considerations for 10 CFR Part 52 goes on to say "...the Commission is not out to secure, single-handedly, the viability of the [nuclear] industry or to shut the general public out. The future of nuclear power depends not only on the licensing process but also on economic trends and events, the safety and reliability of the plants, political fortunes, and much else. The Commission's intent with this rulemaking is to have a sensible and stable procedural framework in place for the consideration of future designs, and to make it possible to resolve safety and environmental issues before plants are built, rather than after."

In February of this year, the Commission directed the staff in COMJSM-00-0003 to assess its technical, licensing, and inspection capabilities and identify enhancements, if any, that would be necessary to ensure that the agency can effectively carry out its responsibilities associated with an early site permit application, a license application and the construction of a new power plant. In addition, the Commission directed the staff to critically assess the regulatory infrastructure supporting both 10 CFR Parts 50 and 52 with particular emphasis on early identification of regulatory issues and potential process improvements. The focus of these efforts is to ensure that the NRC is ready for potential applications for early site permits and new nuclear power plants, certified designs or designs to be certified, and the NRC does not become an impediment should society decide that additional nuclear plants are needed to meet the energy demands of the country. Necessary safety-focused regulations, yes; unnecessary, not safety-focused regulations, no. The staff is working hard to carry out this direction and I am sure you will hear about some of our efforts over the next two days.

Risking being repetitive, I am going to re-start at the beginning. The U.S. Nuclear Regulatory Commission has a three-pronged mandate:

- To protect the common defense and security
- To protect public health and safety, and
- To protect the environment

by the licensing and regulation of peaceful uses of atomic energy. I have long advocated that an adequate and reliable energy supply is an important component of our national security. I firmly believe that our three-pronged mandate is going to endure the test of time because it is good, and it is balanced.

Within that mandate, I am an advocate of change, functioning under the rule of law. As we face the regulatory challenges that are sure to be posed by the certification and licensing of new designs, a series of familiar requirements will have to be met, regardless of the licensing path chosen:

- Public Involvement
- Safety Reviews
- Independent ACRS Review
- Environmental Review
- Public Hearing
- NRC Oversight

I am convinced, by practical experience, that the present pathway for potential licensing success of certified or certifiable new reactor applications is Part 52. First, it exists - not a minor issue; second, it contains the requirements for assurance of safety and the processes for their implementation. And lastly, it can be upgraded to meet technological advances that require new licensing paths, without compromising safety. Windows of opportunity can be opened, yet the price is always the same: reasonable assurance of public health and safety. A new technology, with different design basis phenomenology, e.g., single phase coolant, could present the need for a different pathway. Yet, it would have to face the same requirements listed above. What could be different is the manner in which some of these requirements are addressed. There is definitely room for innovation and improvement, within the safety envelope that has to be provided for assurance of public health and safety.

I am also convinced that the NRC and all stakeholders need to apply common criteria to the tasks at hand. Every success path, however success is defined, should follow these simple criteria: Every path, every step has to be disciplined, meaningful and scrutable.

Allow me to consider widely different roles.

The NRC has the statutory responsibility for conducting licensing and regulation in a predictable, fair, equitable and efficient manner to ensure safety. Every step of the licensing and oversight has to be disciplined, meaningful and scrutable.

Applicants need to satisfy the technical, financial, and marketplace requirements, and meet the NRC and other regulatory requirements. Every step has to be disciplined, meaningful and scrutable.

I have no doubt that there will be objections and opposition and the law of the land will respect them and give them full consideration. The objections will have to be disciplined, meaningful and scrutable.

These common criteria are necessary but they are not sufficient. It is indispensable that what we have learned - and it is much - be incorporated into the science, engineering and technology supporting any new reactors; they have to be as good as the state-of-the-art permits. And so it should be for the regulatory processes. I happen to believe that risk information can be a contributor to disciplined, meaningful and scrutable processes, and to the underlying science and technology.

Someone once wrote a phrase framing how to achieve high performance expectations, and it may be appropriate for this occasion:

Promise... to think only the best,
to work only for the best
and
to expect only the best

Nuclear Power Generation
- Perception and Reality -

	1973 - 1982	2001
Interest Rates	High & Unstable	Low & Stable
Inflation	High & Unstable	Low & Stable
Electrical Demand	Decreasing	Increasing
Socio-political Climate	Negative	Improving
Technical Maturity	Low	High
Regulatory Framework	Low Predictability	High Predictability
Economical Performance	Poor & Unstable	Good & Improving
Environmental Image	Poor	Improving
Safety Image	Poor	Good & Improving
Expectations	Too High	Realistic
Competition/Deregulation	None	High
Standard (certified) Designs	None	Three +
Combined License	No	Yes
Important to National Security	Yes	Yes
Financial Risk	High	Improving
Public Credibility	Low	Good & Improving
Bottom Line	**Low Predictability**	**Good Predictability**

Table 1

Generation IV Initiative

Presentation at ACRS Workshop "Regulatory Challenges for Future Nuclear Power Plants"

June 4, 2001

William D. Magwood IV, Director
Office of Nuclear Energy, Science and Technology

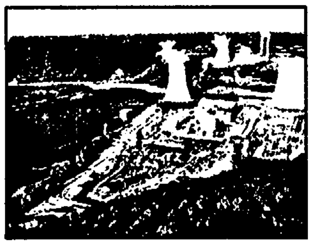
Generation IV Systems

- **Nuclear energy systems deployable by 2030**
- **Systems offering significant advances in**
  - sustainability
  - safety and reliability
  - economics
- **Systems include fuel cycle and power conversion**
- **Diversity of applications (electricity, H2, water, heat)**
- **Deployable in a wide range of markets**

The Evolution of Nuclear Power

Generation I

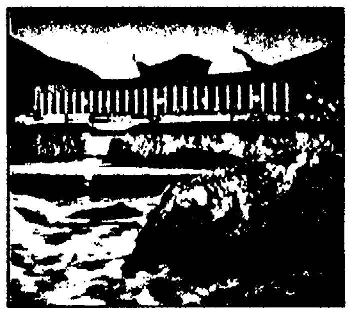
Early Prototype Reactors



- Shippingport
- Dresden, Fermi I
- Magnox

Generation II

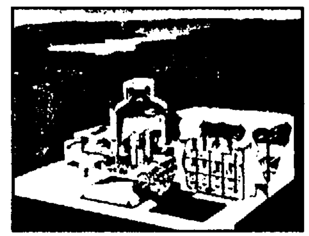
Commercial Power Reactors



- LWR-PWR, BWR
- CANDU
- VVER/RBMK

Generation III

Advanced LWRs



- ABWR
- System 80+
- AP600
- EPR

Near-Term Deployment

Generation III+ Evolutionary Designs Offering Improved Economics

Generation IV

- Highly Economical
- Enhanced Safety
- Minimal Waste
- Proliferation Resistant

Gen I	Gen II	Gen III	Gen III+	Gen IV

1950 — 1960 — 1970 — 1980 — 1990 — 2000 — 2010 — 2020 — 2030

Nuclear Energy Research Advisory Committee (NERAC)

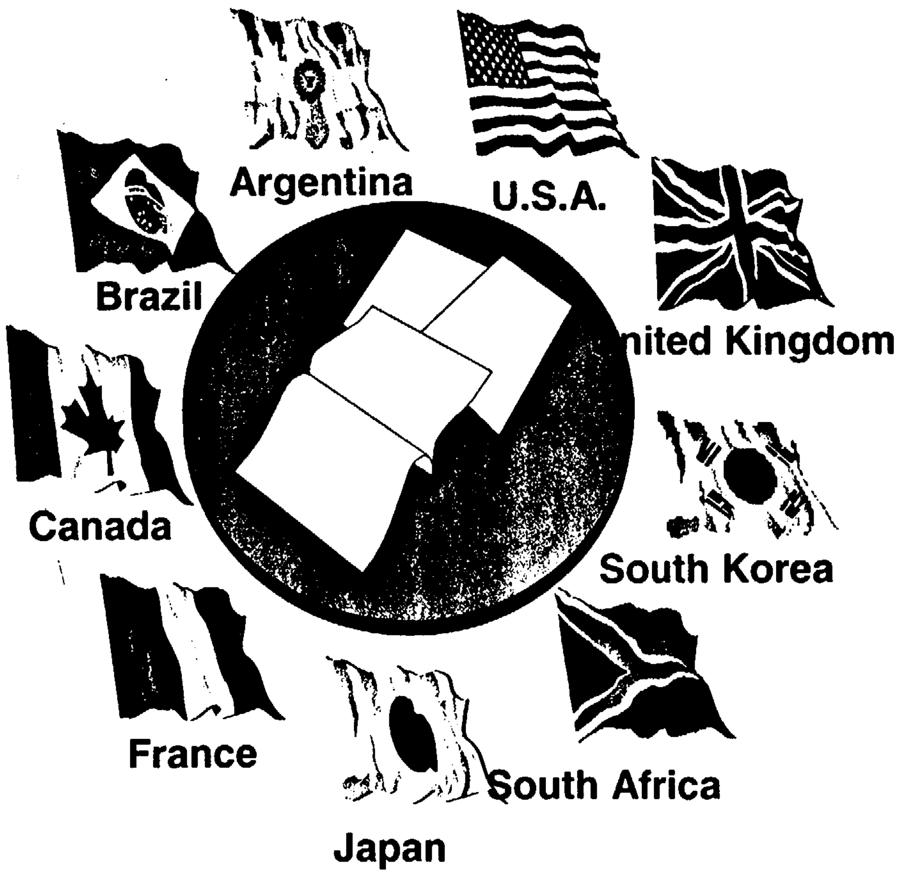
- **Subcommittee on Generation IV Technology Planning**
  - Established in October 2000 to provide guidance on development of the Generation IV Technology Roadmap
  - Membership from U. S. Industry, laboratories, and academia
  - Co-chaired by Neil Todreas, MIT and Sal Levy, GE (retired)

Nuclear Energy Research Advisory Committee (NERAC)

Subcommittee Charter: Gen IV Technology Roadmap

- Establish goals that define the requirements for Generation IV nuclear energy plants
- Suggest paths forward to resolve technical and institutional issues for Near-Term Deployment (by 2010)
- Recommend Gen IV R&D Plan
  - Sequencing of R&D task and initial cost estimates
  - National and international collaboration
  - Systems must be deployable by 2030

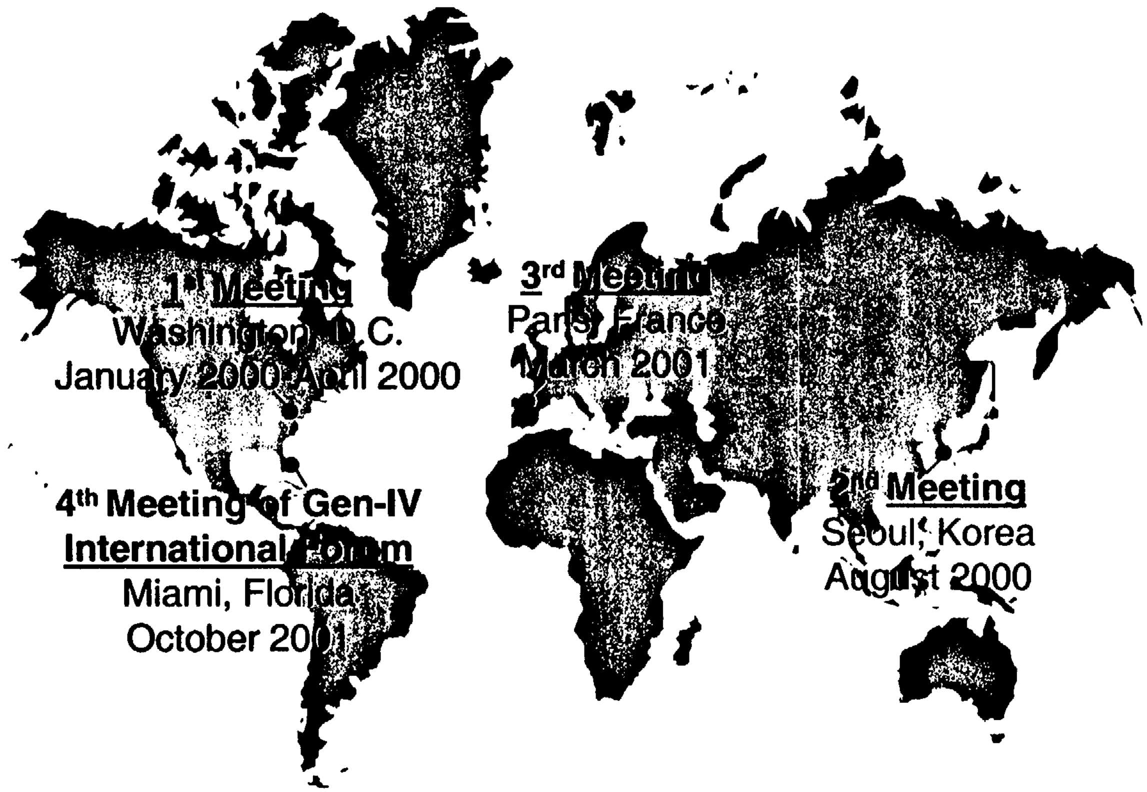
Generation IV International Forum



- Facilitate research planning and international cooperation between countries interested in the future of Nuclear Energy
- Led by Policy Committee, composed of senior nuclear technology official representing member governments
- Observers from:
  - International Atomic Energy Agency
  - OECD/Nuclear Energy Agency
  - European Commission
  - U.S. Nuclear Regulatory Commission
  - U.S. Department of State

Generation IV International Forum

- Endorsed Gen-IV technology goals
- Internationalized the Gen-IV Technology Roadmap effort
- Finalized charter governing memberships and objectives



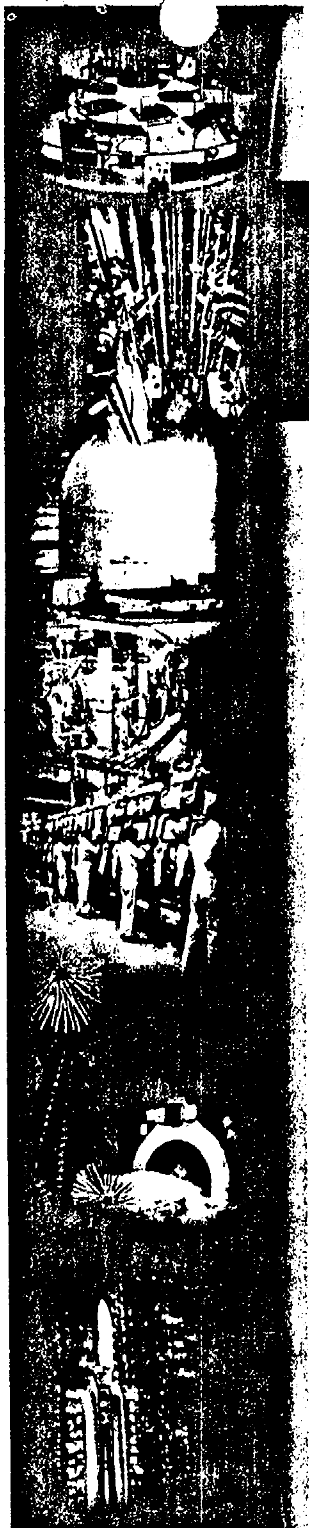
Generation IV Initiative

Near-term Objectives

- Establish Near-term Deployment Working Group
- Identify institutional and regulatory barriers to new plant deployment in the U.S.
- Provide recommendations on appropriate government actions to assist in addressing barriers (complete by September 2001)

Long-term Objectives

- Establish Gen-IV Technology Project
- Identify and evaluate most promising nuclear energy system concepts
- Provide comprehensive R&D plan to support future commercialization of the best concepts (complete by September 2002)



Generation IV Goals and Roadmap Effort

Presentation at ACRS Workshop "Regulatory Challenges for Future Nuclear Power Plants"

June 4, 2001

Dr. Rob M. Versluis

Office of Technology and International Cooperation

Generation IV Technology Roadmap

- **Identify and evaluate most promising nuclear energy system concepts (Oct '00 - Sep '02)**
- **Advisory group: Generation IV Roadmap NERAC Subcommittee (GRNS)**
- **Working Groups:**
  - ~50 U.S. experts from industry, labs, academia
  - ~40 experts from Generation IV International Forum (GIF) member countries & organizations
- **R&D Plan to support future commercialization of the best concepts**

Generation IV Technology Roadmap: Goals

Goals

- **Reflect mid-century vision of energy needs (2030)**
- **Provide basis for evaluating nuclear energy systems and identify the most promising concepts**

Sustainability Goals
- *Resource inputs*
- *Waste outputs*
- *Nonproliferation*

Safety & Reliability Goals
- *Excellence*
- *Core damage*
- *Emergency response*

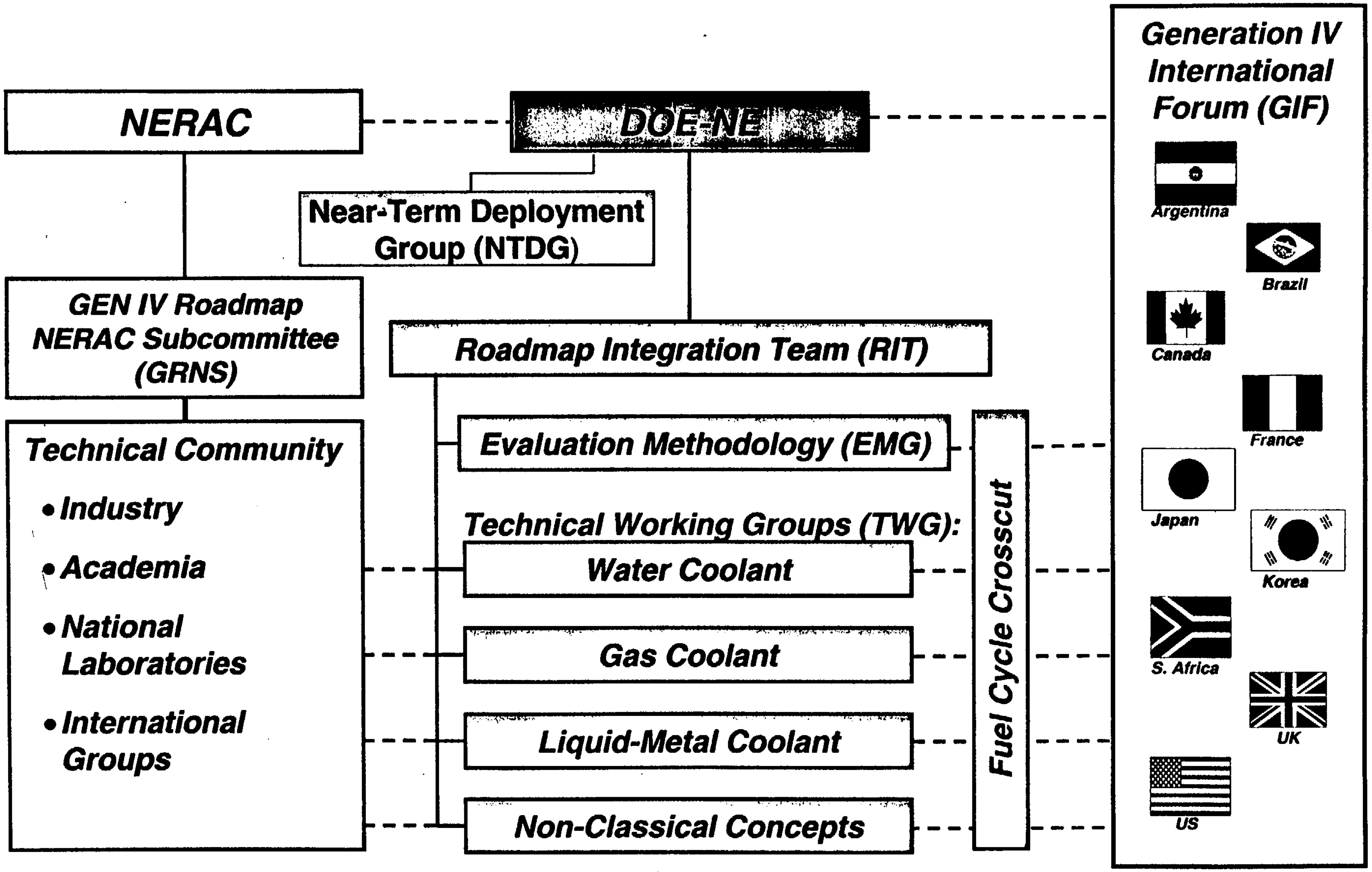
Economics Goals
- *Life cycle cost*
- *Risk to capital*

Key Definition: System

Generation IV System:

- **An entire energy production system, including**
  - nuclear fuel cycle front and back end
  - nuclear reactor
  - power conversion equipment and its connection to the distribution system
  - electricity, hydrogen, fresh water, process heat, district heat, propulsion
  - infrastructure for manufacture and deployment of the plant
- **Limited to systems that are likely to be commercially viable by 2030**
- **Primary energy generators based on critical fission reactors**

Generation IV Technology Roadmap: Organization



Schedule for Producing the Roadmap

Four Phases over Two Years:

Phase I: Initial work

Oct '00 – Jan '01 – Completed

Phase II: Needs assessment

Jan '01 – Jan '02 – Jan '02 Draft Roadmap

Phase III: Response development

Oct '01 – May '02 – May '02 Interim Roadmap

Phase IV: Implementation planning

May '02 – Sep '02 – Sep '02 Final Roadmap

First Steps: Goals and Plans

Derive technology goals based on industry needs

- Goals have been drafted by GRNS and GIF
- Captured in Technology Goals Document

Plan the activity

- Roadmap Development Guide drafted by RIT
- Working groups have been convened including international participation

Determine how to measure concepts against goals

- Develop criteria and metrics for each goal
- Continue on to develop evaluation methodology
- Conducted by EMG, with the RIT and GRNS

Next Steps: Concepts

Identify concepts for evaluation

- Drawn from a broad international base
- Concepts adopted or synthesized by TWGs
- Concepts grouped into “concept sets”

Detail the most promising concepts

- Interactions between TWGs & concept teams/advocates
- Active study and comparison of underlying technology
- “Screening for Potential” guided by EMG criteria
- Evaluations guided by EMG metrics

Key Definition: Concepts

Concept:

A technical approach for a Gen IV system with enough detail to allow evaluation against the goals, but broad enough to allow for optional features and trades.

Concept Set:

A logical grouping of concepts that are similar enough to allow their common evaluation.

The Second Year: Evaluate & Assemble

Evaluate the most viable concepts

- Compare concept performance to goals
- Identify technology gaps
- TWGs lead – RIT/EMG reviews – DOE approves – GIF endorses

Assemble Roadmap to support the most promising concepts

- Identify R&D needed to close gaps in areas of crosscutting technology
- Assemble a program plan with recommended phases
- Groups report – RIT integrates – DOE approves – GIF endorses

Planned Evaluation Stages

- *Request for information* *March 2001*
  Concept elicitation, sorting, and characterization
- *Screening for Potential* *July 2001*
  Concept studies
  (assessment of technical needs by concept)
- *Final screening* *April 2002*
  R&D plan development
- *Roadmap completion* *September 2002*

Viability R&D

- *First down-selection*
  Performance R&D (industry participation)
- *Second down-selection*
  Demonstration w/industry, design, regulatory reviews

Backups

Technology Working Groups 1–4

Charter

- Identify Gen IV concepts for evaluation, evaluate their potential against the goals, their technology gaps and needs, and recommended R&D priority.

Special Features

- Groups will author major sections of the roadmap on concepts, technology gaps and R&D needs
- Group members will staff the crosscut groups in the second year

Evaluation Methodology Group

Charter

- Develop a process for the systematic evaluation of the comparative performance of proposed Gen IV concepts against the established Gen IV goals.

Special Features

- Early delivery of products in Feb/Mar and May 2001
- Continued refinement of methodology
- Review of the TWG analyses to assure a consistent approach

Fuel Cycle Crosscut Group

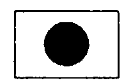
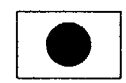
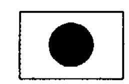
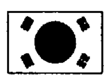
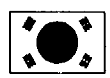
Charter

- Examine fuel resource input and waste output from a survey of Generation IV fuel cycles, consistent with projected energy demand scenarios. The survey of fuel cycles will include currently deployed and proposed fuel cycles.

Special Features

- Members mostly drawn from the TWGs and EMG
- 8–10 month time frame for delivery of products

International Participation in Generation IV Roadmap

	Water	Gas	Liquid Metal	Non-Classical	Eval. Methods	Fuel cycle
Argentina						
Brazil						
Canada						
France						
Japan						
Korea						
South Africa						
United Kingdom						
United States						

Department of Energy Near-Term Deployment Working Group

Presentation at the ACRS Workshop - Regulatory Challenges in the Licensing of Generation 3+ and Generation 4 Reactors

Thomas P. Miller

June 4, 2001

Near-Term Deployment Group

- **Mission - Identify the technical, institutional and regulatory gaps to the near term deployment of new nuclear plants and recommend actions that should be taken by DOE.**
  - Orders by 2005
  - Multiple plants in commercial operation by 2010
- **Participants - multi-disciplined nuclear industry group**
  - Nuclear Utilities - Duke, Southern Nuclear, Exelon
  - Reactor Vendors - Westinghouse, General Electric, General Atomics
  - National Laboratories - ANL, INEEL
  - Academia - Penn State
  - Industry - EPRI
  - Government - DOE-NE
  - NERAC

Near-Term Deployment Group

Deliverables

- Near-Term Actions for New Plant Deployment
- Near-Term Deployment Report (Roadmap)

Near-Term Actions For New Plant Deployment

- Overview of recommended DOE activities and FY 02/03 funding needs
- Intended for use during DOE budget hearing process and DOE-NE input to VP Energy Task Force
- Presented to NEI and New Plant Task Force
- Significant Activities include:
  - Early Site Permit Demonstration (10CFR52)
  - Combined Construction/Operating License (COL)Demonstration (10CFR52)
  - Design Certification of 1000+ MWe ALWR
  - Confirmatory Testing and Code Validation of Advanced Reactor Utilizing New Technology

Near-Term Deployment Group

Near-Term Deployment Report

- To be Issued by September 30, 2001
- Based on evaluation of industry response to RFI

Request for Information (RFI)

- Issued April 4, 2001 to reactor designers, AEs, nuclear plant owners/operators, Gen IV participants, and other stakeholders
- Issued to NEI New Plant Task Force members
- Public notice through Commerce Business Daily (CBD)
- Solicits identification of design-specific, site-related and generic barriers to deployment of new nuclear plants by 2010
- Responses due May 4, 2001- received responses from 12 organizations
- RFI response under review

Near-Term Deployment Group

RFI requested information in two areas:

- **Specific Deployment Candidate Designs that meet six criteria**
  - Credible plan for gaining regulatory acceptance
  - Existence of industrial infrastructure
  - Credible plan for commercialization
  - Cost-sharing between industry and government
  - Demonstration of economic competitiveness
  - Reliance on existing fuel cycle structure
- **Generic & Design Specific Gaps**
  - Known gaps provided requiring ranking and possible solutions
  - Other gaps to be identified by respondent

Near-Term Deployment Group

Design Specific Responses

SW 1000	Framatome
PBMR	Exelon/PBMR
AP600/AP1000	Westinghouse
IRIS	Westinghouse
GT-MHR	General Atomics
ABWR	General Electric

Near-Term Deployment Group

Generic Gaps Responses

- ESP Demonstration
- COL Demonstration
- Construction Inspection & ITACC
- Risk-Informed Regulation for Future Design Certifications
  - Emergency Planning and Plant Security
- Advanced Fabrication, Modularization and Construction Technologies,
- Standardized Life-Cycle Information & Configuration Control Systems
- High Level Waste Disposal Resolution
- Risk Management Tool
- Public Influence and Acceptance
- Appropriate Resource and Financial Arrangements

Generation IV Concepts

Presentation at ACRS Workshop "Regulatory Challenges for Future Nuclear Power Plants"

June 4, 2001

Dr. Rob M. Versluis
Office of Technology and International Cooperation

Overview

- Request for concept information (RFI)
- RFI response
- Concept statistics & key features
- Grouping of concepts
- Current activities on concept evaluation

Key Definition: Concepts

Concept:

A technical approach for a Gen IV system with enough detail to allow evaluation against the goals, but broad enough to allow for optional features and trades.

Concept Set:

A logical grouping of concepts that are similar enough to allow their common evaluation.

Concept Statistics (5/18/2001)

Total: 94

By reactor coolant type

- Water 28
- Gas 17
- Liquid Metal 32
- Non-classical 17

By organization type

- University 27
- Industry 22
- Laboratory 45

By country

- France 3
- Japan 19
- Korea 10
- UK 4
- US 45
- 7 Others* 13

*Argentina, Brazil, Canada, Germany, Italy, Netherlands, Russian Federation

Concepts with Water Coolant

Variables

- Coolant (H_2O, D_2O)
- Coolant phase & conditions
- Spectrum (thermal, epi-thermal, fast)
- Primary system layout (conventional, integral)
- Fuel cycle (U vs.Th, once-through vs. recycle)
- Thermal output
- Maturity

Concepts with Water Coolant (cont)

Crosscutting R&D Issues

- High temperature materials
- Modular manufacturing technologies
- Internal control rods
- I&C

Concepts with Gas Coolant

Variables

- **Reactor concepts**
  - ➢ GT-MHR
  - ➢ PBMR
  - ➢ Fluidized Bed Reactor
  - ➢ GCFR
- **Applications of fission heat**
  - ➢ Electricity generation: direct vs. indirect cycle
  - ➢ Process heat applications (industrial smelting, petroleum refining, hydrocarbon reforming, coal conversion, etc.)
  - ➢ Desalination

Concepts with Gas Coolant (cont'd)

- **Fuel forms and fuel cycles**
  - LEU
  - Thorium
  - U-Pu

- **Generic R&D issues**
  - Fuel fabrication quality assurance
  - Fuel performance -- integrity and FP retention
  - Lifetime temperature and irradiation behavior of graphite structure
  - High temperature materials and equipment
  - Passive decay heat removal for fast-spectrum concepts

Concepts with Liquid Metal Coolant

- **Variables**
  - ➢ Size (large/monolithic, modular, transportable) and targeted clients
  - ➢ Coolant (Na, Pb-alloy, Pb, ...)
  - ➢ Fuel type (oxide, metal, nitride, composites)
  - ➢ Primary system layout (loop, pool)
  - ➢ BOP options and energy products
  - ➢ Energy conversion options
  - ➢ Fuel recycle technology (aqueous, dry)

Non-Classical Concepts

- **Focus: adequately defined concepts with significant potential**
- **Variables**
  - Cooling approach (convection, conduction, radiation)
  - Coolant (molten salt, organic coolant)
  - Fuel phase (solid, liquid, gas/vapor)
  - Electricity generation technology conversion (turbine, gas MHD, direct conversion of fission-fragment energy)
  - Alternative energy products or services
  - Fuel cycle

Non-Classical Concepts (cont)

- **Crosscut issues**
  - Modular deployable
  - Hydrogen production and very high temperature systems
  - Advanced fuels and fuel management techniques
  - Energy conversion systems (esp. non-Rankine)

Concept Grouping

- **TWG's have grouped concepts into "concept sets"**

- **Concept sets share**
  - ➢ Technology base
  - ➢ Design approach

- **Rationale for grouping**
  - ➢ Efficient division of TWG analysis effort
  - ➢ Streamline evaluation process
  - ➢ Avoid premature down-selection

Concept Grouping: Water TWG

• PWR loop reactors	(3)
• Integral primary system PWR's	(6)
• Integral BWRs	(6)
• Pressure tube reactors	(3)
• High conversion cores	(11)
• Supercritical water reactors	(3)
• Advanced fuel cycle concepts	(14)

- MOX
- Thorium
- DUPIC
- Marble Fuel
- Neptunium

Concept Grouping: Gas TWG

- **Pebble bed modular reactor concepts (5)**
- **Prismatic modular reactor concepts (5)**
- **Very high temperature (~1500°C) reactor (1)**
- **Fast-spectrum reactor concepts (5)**
- **Others (4)**
  - ➢ Fluidized bed
  - ➢ Moving ignition zone concepts

Concept Grouping: Liquid Metal TWG

- **Four major categories of concepts:**
  - ➢ Medium-to-large oxide-fueled systems (6)
  - ➢ Medium-sized metal-fueled systems (8)
  - ➢ Medium-sized Pb/Pb-Bi systems (8)
  - ➢ Small-sized Pb/Pb-Bi systems (6)
- **Liquid Metal TWG is also examining three supporting technology areas**
  - ➢ Fuels (oxide, metal, nitride)
  - ➢ Coolants (Na, Pb/Pb-Bi)
  - ➢ Fuel Cycle (advanced aqueous, pyroprocess)

Concept Grouping: Non-Classical Systems TWG

- Eutectic metallic fuel (2)
- Molten salt fuel (4)
- Gas core reactor (1)
- Molten salt cooled/solid fuel (1)
- Organic cooled reactor (1)
- Solid conduction/heat pipe (1)
- Fission product direct energy conversion (2)

Current Activities

- **TWG's are analyzing the candidate concepts for**
  - ➢ Performance potential relative to the technology goals
  - ➢ Technology gaps

- **A report will be prepared this fiscal year describing**
  - ➢ Concepts
  - ➢ R&D needs
  - ➢ Results of the initial "screening for potential" evaluations