What that means, that the inspection intervals have to be long. And you see inspection effective in that case. The other issue that comes up is many components in advanced reactors are not excessive. For example, containing vessels.

So there is also an additional problem with accessibility. If I can't inspect important components, what good would periodic service inspections do us. So there is some issues related to those two areas.

MEMBER FORD: So that last one really refers to inspection intervals, not looking at PIE or --

MR. MUSCARA: In-service inspection for the presence of fluence.

MEMBER FORD: Okay.

CHAIRMAN KRESS: But see, the IRIS has a lifetime of 8 years for the cooler or something like that?

MR. MUSCARA: Eight. It's got all the components. but really, it's a challenging inspection problem there to address this. In the area of design codes from the telecomponents, there is a general lack of design codes and standards. We do have available ASME code case N-499, and N-201 and there is a fairly new subsection NH for application to high temperature components.

Well these codes were developed based on data from the '70s and '80s from the LMFBR area. That means a lot of the data that has gone into these codes is taken in air and/or sodium.

In addition, data from the '90s have come up with better correlations for relating creep and the creep-fatigue interaction, which is not addressed in the code.

CHAIRMAN KRESS: Do you or some of the NRC people serve on -- people putting together these coded?

MR. MUSCARA: Yes, we participate in several committees. The ASME, for example, is now beginning to think about what needs there are for the future for these advanced reactors. I have had a meeting with standards development organizations. And describing the need for codes and standards in different areas related to materials and inspection. And in fact, I was able to get some work started, which I can cover a little bit later. But right now, I think the codes and standards committees are lagging behind on doing any work in this area. And what is in place, I believe it is not appropriate for the high temperature gas cooled reactors.

MEMBER SIEBER: What pressure does the pebble bed operate at in the primary circuit? It is not real high?

MR. MUSCARA: No it is much more like the boiling water reactor.

MEMBER SIEBER: Right.

MEMBER ROSEN: It's that high?

MEMBER SIEBER: Yeah.

CHAIRMAN KRESS: The helium was not a good heat to get the heat transfer.

MEMBER SIEBER: That's sort of an advantage. Because you don't have quite the stresses in the vessels and the various compounds that you would if you were operating at perhaps double that pressure. But the temperature is way up there.

MR. MUSCARA: Yes. And a key lack within the codes is of course the inclusion of the effects of the environment in the design, both for fatigue and for creep. And the experience we have had with light water reactors tell us that the effects of environment are quite important.

You know, when we designed and built the light water reactors, we had high purity water and therefore didn't worry too much about things like parts per million of impurities. But those are the kinds of things that will really get us into trouble.

When discussing the environmental effects on fatigue, creep, and stress corrosion cracking, as I have mentioned, there is a lack of data for fatigue, for creep, and for stress corrosion cracking for evaluating the lifetime design of these components.

We know that temperature stress, strain rate, strain amplitudes and impurities such as oxygen and chloride, reduce the fatigue in creep life and increases susceptibility to stress corrosion and cracking.

In addition, you get an increase in crack growth rates due to the effects of the impurities the environment. Therefore research is needed on fatigue, creep, stress corrosion cracking and crevice corrosion cracking to take into account the effects of oxygen, chloride, temperature strain, strain rate, strain range.

The results of this research will help us to quantify and confirm if these degradation mechanisms do -- for the helium environment. And if they do play a major role, then we would have a data base for updating the current codes and standards.

MEMBER WALLIS: It seems amazing that you have to do this research?

MR. MUSCARA: Again, --

MEMBER WALLIS: -- a state-of-the-art here somehow. Why should the NRC do that?

MEMBER SIEBER: Well there isn't any art, right? In this kind of application. So somebody has to.

MEMBER WALLIS: What business do people have designing something if they don't understand fatigue, creep and --

MR. MUSCARA: This is a policy question they have sent up on to the Commission. Can we design and license these plants when these are not adequate codes for designing them. And in my view, the effects of environment are not taken into account we miscalculate.

MEMBER WALLIS: Why should NRC do it?

MR. MUSCARA: It is much like we discussed this morning, this is work that needs to be done.

MEMBER WALLIS: So it even seems worse than this morning. This fatigue, creep and corrosion cracking of materials is a very basic thing throughout the industry.

MR. MUSCARA: Yes it is. And I think when we designed the light waster reactors, it was fairly basic then also, but we accepted. For example in fatigue, data was developed on small specimens, or smooth specimens, polished surfaces tested in air. And then we found that in fact if you test the same specimen, even though it is polished and small, there are 70 times the effect of the effect of oxygen and water. So the life could be will be by a factor of 70 times different than what we designed those plants and accepted them.

So my concern is we did it then. And I am trying to make use of lessons learned from the light water reactors and bring up these issues.

MEMBER ROSEN: From a first principles basis, why should we be surprised with that result?

MR. MUSCARA: At this stage, we should not be surprised. I mean we have seen this happening with light water reactors. But the point is, that the work hasn't been done.

I have seen some work where the effects of environment were trying to be addressed, but unfortunately the most important parameters, oxygen and chloride were not included in the impurity environment. So there is some data that is limited and does not address the key parameters. So it is work that needs to be done. I think the work needs to be done and considered while we are reviewing these license applications.

MEMBER SIEBER: I think also the light water reactor data -- much lower temperature, a third of the temperature. And so the data that is available is out of range. I mean it doesn't include even the operating condition.

MR. MUSCARA: Some of the components are higher temperature. And in fact some components exhibit creep which we don't see in the light water reactor. And in creep also, there is a factor of impurities.

MEMBER SIEBER: Is there an opportunity to use codes and standards from the aircraft industry? You know jet engines operate at pretty high temperatures in the same way as combustion turbines?

MR. MUSCARA: That is true I think from a design, I think for the process, it may be quite adequate from the data point of view. I am not sure that the data is --

MEMBER SIEBER: Of course if you take a jet engine from an airplane and you look at its service life between overhauls, you couldn't afford to run a power plant like that.

MR. FLACK: But again, just to reemphasize the fact is that the plan doesn't say that NRC will do the research. I mean we are seeking it out through international cooperations, collaborations, and industry as well as what we may have to do ourselves. So it all has to be determined.

MR. MUSCARA: But the fact is that is a key area. The data is not there, we need to get going soon to get the data. For example, we have done the research in the light water reactor area. It wasn't the industry that came up and said, you know we have an effect of the environment it was NRC work that discovered this effect.

MEMBER SIEBER: Right.

MEMBER WALLIS: This is a research plan for the NRC. This is not a research plan for industry, I take it.

MR. MUSCARA: When we developed the plan, the general philosophy was to identify key areas that needed to be addressed.

MEMBER ROSEN: And that discussion will go on between NRR and the licensees -- the applicants.

MR. FLACK: That's right.

MEMBER ROSEN: As to how it is going to get done. And if the answer comes back: NRC you do it all, then the answer is fine. We will do it all in

2090.

MEMBER SIEBER: Well either that or give us a charge card, right?

MR. FLACK: Or a containment.

MEMBER ROSEN: It's a fair question. If the industry wants the NRC to do it all, the NRC should get to define the schedule. The industry might not like the schedule.

MEMBER FORD: But just to interrupt for a minute Joe. We are all saying that and I can understand why you are all saying that. Is it a responsible position to be though? Should we not be in the position of being an informed regulator? And i.e, have the answers to a certain extent in our pocket?

It is a question. I don't know the answer to the question, is the question.

MEMBER SIEBER: I think that we are obligated to be an informed regulator. On the other hand, if you aren't informed on even a given area, you either come up with an alternative or defense-in-depth or don't approve it. And that is up to the industry to take one of those alternatives.

One way to deal with the high temperature in creep problems is to say, here is the maximum temperature that we are going to allow you to operate this at. And when the efficiency goes to pot when you do that. And they say well it is not worth building it. You know there are all kinds of decisions that can be made and I think that --

MR. MUSCARA: But even if we say that, we have to have some basis for it. For example, I don't want to base it on information we have on error data. I would like to base that decision on what happens to these components in the actual environment.

MEMBER SIEBER: I think that is true.

MEMBER WALLIS: But you could ask them to do that. Evaluating the lifetime design is the responsibility of the designer. Isn't it? Primarily, and then you have to check it.

MR. MUSCARA: We need to --

MEMBER WALLIS: We happen to have the primary responsibility.

MR. MUSCARA: And the contention these days is that helium is an earth and it is pure, therefore data in air or helium is acceptable and adequate. Our experience tells us that it might not be the case. So some of this research may fall into an area that we call anticipatory research. If the plan is designed and built, I don't expect a problem the first year. But I might expect it after ten years.

MEMBER SIEBER: It's as pure as primary grade water.

MR. MUSCARA: Right, that's the point. It is pure -- earth quotation marks. We have three parts per million oxygen in the high temperature gas cooled environment.

MEMBER SIEBER: I imagine in these compressors and turbines you have to have some kind of lubrication which introduces. That is a major source of all these impurities. Because there are bearings in there that are usually pretty high speed devices.

MR. MUSCARA: There is, at least for the pebble bed, there's a purification system. But when I've looked at the information from the AVR, what goes into the system comes back out. With respect to oxygen for example, it comes out at less than a part per million oxygen. But it goes in at 3 parts per million. So during the cycle it picks up oxygen enough to cause the degradation of materials.

MEMBER SIEBER: And everything ahead of the purification unit, you know up stream, is exposed to the three parts per million.

MR. MUSCARA: Right. So the connecting

pipe is an issue. What happens that these high temperature gas cooled reactors, the connecting pipe has been designed, fabricated and inspected to the same rules as a vessel. So the contention is that the pipe, therefore is a vessel. And we consider a vessel, while doing that, then there is no double ended break as a design basis.

And therefore there are no mitigating systems incorporated into the design. Now, in a pipe as a vessel, it is not really realistic. Even though the pipe is built constructed, and inspected same as vessel, because of the diameter, the vessel itself is much, much thicker for the same working pressure than the pipe. So should a degradation mechanism occur in the pipe, expected or unexpected, it goes through the walls relatively quickly.

And therefore even a vessel, except for some recent experience, you don't expect degradation mechanism go through the vessel in short periods of time have a chance to be -- by inspection, etc.

So I think it is quite a major difference between the pipe and the vessel. You can inspect it the same way, we can build it the same way, but it is much thinner. That is a fact, if you want to build this thing six inches thick, then fine. Then they can call it a vessel. But it is not, it is less than two inches thick, it is very much like --

MEMBER ROSEN: Aren't current day piping, primary piping designed, fabricated and inspected to the same rules as the RPV?

MR. MUSCARA: Precisely.

MEMBER ROSEN: So when we don't allow that in LWR, so what changed is what I am asking?

MR. MUSCARA: Right, we have had the contention from the industry that they are built that way. And therefore the probability of failure is very low. And I am saying wait a minute. What about all the experience? These pipes do crack. They have cracked.

MEMBER ROSEN: But there arguments just saying that we are designing and fabricating and inspecting the same rules as the RPV, therefore, that we don't have to do anything different, it doesn't hold water on the surface. Because that is what we are doing already for light water reactors and we do take double ended breaks.

MR. MUSCARA: It is very much the same process for design, fabricating, and inspecting you know the primary system components.

MEMBER SIEBER: But the piping code is

different than the pressure vessel codes.

MR. MUSCARA: The design.

MEMBER SIEBER: So it is not exactly the same. There are some things that are different, but you are right. The smaller the diameter of the pipe, the thinner the wall could be. Look at the steam generator tube, it is very thin. And you can crack through one of them pretty fast.

MR. MUSCARA: -- we are not really planning necessarily any research on this, but we will be making use of the research in the other areas to try and determine what is the potential, what's the probability of failure in this pipe. So if we bring it up as an issue, and the research we will be conducting on fatigue and creep and environmental effects, should apply to the analysis of this pipe, how clever is it that this thing is not that, the probability is very, very low.

MEMBER FORD: So, I am just trying to follow up on the decision that came earlier and that statement you just made. So the objective of this and the other work is to come up with what do we know currently and what is necessary to be done in order to find the probability of failure of the component. That would then lead into a higher level risk informed basis, the probability of our CDF or LERF appropriate higher level safety might actually cost. Is that the reason?

MEMBER SIEBER: Well I looked at it a little bit differently. A licensee is going to come in and they are going to make an assertion. And the staff is going to ask the licensee, prove to me that your assertion is correct. And the staff has to have enough data and knowledge to be able to make that judgement.

And so, you end up with both the industry, the vendors doing some work to assert their end of the argument. Staff has got to be knowledgeable enough and have at its own command, sufficient data and experience to say you are right or you are wrong. And that is how I see this coming out.

MR. FLACK: Exactly. And that could end up being the difference between one kind of accident versus another kind of accident. And what you have to design the rest of the facility to withstand.

MEMBER FORD: But from your research, is to tell the licensee, prove to me the probability of the failure of this component by whatever mechanism is less than such and such. Is that the --

MR. MUSCARA: In my mind, that is a key

aspect also because both the design and the licensing these plans is moving more and more towards risk informed and risk based. And you need to have reasonably data to conduct these evaluations.

And since there is a lack of experience with these materials and components, we would have to predict it through some probabilistic failure mechanics. To do that you must identify degradation mechanisms, initiation times, the growth rates and so on.

MEMBER SIEBER: That's right, and the output is going to be a distribution.

MR. MUSCARA: Yes.

MEMBER SIEBER: So you can define the uncertainty and all of these get factored into this grand equation that says here is the risk of this facility.

MEMBER FORD: Yes, but the proof of the pudding, that licensee can maintain that low level of risk. That is his responsibility. And you have got to be in the position of being an informed regulator to understand that he is not pulling the wool over your eyes.

MEMBER SIEBER: Well it goes beyond that a little bit. The American people look to the agency to keep them safe within the parameters and the safety goals that they have set.

So if one of these plants goes down the tubes, licensee of course will feel some financial heat and regulatory heat. But the agency itself will feel the ire of the population whom we are sworn to protect. So it goes both ways.

MR. MUSCARA: So I think with the connecting pipe issue, essentially because it is designed as a vessel, doesn't really make much sense, number one. Number two, if you are going to design it without assuming double-ended break, you have to show that probability failure is very, very, very small. And I don't think you can do that without the information that we are hoping we can generate.

MEMBER ROSEN: Where does leak-before-break come into this discussion or doesn't it?

MR. MUSCARA: I hadn't planned on discussing it.

MEMBER ROSEN: Well, isn't it part of the discussion on this connecting pipe? If you have to assume that the pipe is a pipe, not a vessel, then can you assume that in the size range that that is going to be used, that the pipe is likely to leak in a detectable way before it breaks? And if you assume that, which I think they might justifiably try to argue. What degree of inspection would we require for leak-before-break in order to limit the break size. Maybe some pipes could be excluded as there is now being discussed in light water reactor family. While others can't.

MR. MUSCARA: Right. In general, we look at is there a potential for degradation of mechanism before we allow the leak-before-break. Because of potential for degradation mechanism, we don't allow it. And in this case, I don't see the data that is showing us, that for example, 800 age, is not susceptible to degradation in the impurity requirement of the helium gas.

MEMBER ROSEN: That is the answer I expected you to give. So we have to show that there is no degradation mechanism. When we are dealing with high temperature piping for which there is no experience it can't show.

MR. MUSCARA: And the -- light water reactor.

MEMBER ROSEN: Sure, and you have enumerated a lot of potential ones.

MEMBER BONACA: One thing that I wanted to point out. You say the corrosion in the lined base is limited in the system as incorporated. So the cracking occurs in the welds anyway irrespective of the way you build this pipe. How can the contention be made. I mean still, you have a concern with cracking through the weld, right?

MR. MUSCARA: Right. And that of course that's been the issue of sensitization over the piping in high residual stresses in that zone. With a different material, it may be more sensitized in the welding. But the other effects may be there during the operation.

MEMBER BONACA: All right, so still, even if you had capability of a vessel, that is an issue of how you put together this components in a way that you would not have potentially a break into the welds.

MEMBER ROSEN: These pipes are cooling down from that to ambient temperature from much higher temperatures than they are typically in light water reactors. I mean they go to operating temperature and when you cool them down, they come to ambient temperatures. A much bigger temperature swing much higher fatigue line.

MR. MUSCARA: Yes, depending on the design and where the insulation is placed. In the one case, the insulation is inside the pipe. In other cases it is outside the pipe. So if it is outside the pipe, you do get some bigger --.

MEMBER ROSEN: Insulation inside the pipe?

MR. MUSCARA: Yes. I think the pebble bed had their insulation jackets inside the duct pipe. In other design, insulation is on the outside. I may get the two mixed up, the GA versus --

MEMBER SIEBER: I think one of the problems was leak-before-break in a gas reactor is your ability to detect the leak. In a water reactor there is a puddle on the floor. Or humidity in the room, but here all you have is your voice gets a little higher when you go into the enclosure.

CHAIRMAN KRESS: There's a possibility of casing emissions that you can hear.

MEMBER SIEBER: Possibility.

CHAIRMAN KRESS: Well they --

MEMBER SIEBER: Some people claim that really works as well.

MR. MUSCARA: In the area of carburization, decarburization and oxidation, these phenomena are dependent on the composition of the coolant. And of course the presence of graphite particles.

Carburization in ferretic steels will lead to a hard brittle surface where cementite is formed at the surface.

For austenitics we would get carbide, chromium carbide formation at the expense of depleting the chromium. So you could leave the surface of the stainless susceptible to cracking.

Decarburization on the other hand takes the carbon away from the materials. So it leaves a softer surface layer and reduced fatigue and creep swing.

So we would need to study these phenomena as a function of time, temperature and in helium gas with impurities including the oxygen. One would conduct metallographic studies to determine whether these reactions have taken place. And also mechanical testing to determine the degree to which the strength has been reduced.

And your objective with research of course would be to characterize and bound the conditions under which the phenomena occur. As I mentioned a little bit earlier, this is going to be a very close balance between being a reducing atmosphere and an oxidizing atmosphere.

For example, I asked a question both in China and Japan about had they thought about carburization in their high temperature helium reactors. And the response from Japan, was yes they had. And in fact, they inject a little oxygen to maintain an oxidizing atmosphere to avoid carburization. Which is great for carburization, but now you are leaving susceptible to corrosion and stress corrosion cracking.

So with the experience with light water reactors and steam generators, there has been a very fine balance there also. Anytime you solve the problems with steam generators, we create another corrosion problem.

And so the conditions under which these things happen haven't really been defined very well. And I think part of the objectives we are trying find these conditions to know when to expect carburization, Decarburization and oxidation.

MEMBER ROSEN: How does decarburization proceed?

MR. MUSCARA: Decarburization? Just the activity of the carbon and the gas versus the carbon in the steel. It is lower in the gas, so that carbon diffuses out of the steel into the gas. And leaves a very soft material, very much like an iron instead of a steel.

MEMBER SIEBER: But that's a surface effect, is it not, pretty much?

MR. MUSCARA: Yes. Those would be surface effects. And what would happen because of the different properties in the surface layer, both the strength and thermal, that during operation you create stresses in the newer surface area. You could initiate cracking and then of course propagating a little bit different and a lot easier.

MEMBER SIEBER: Okay, thank you.

MR. MUSCARA: Well the issue of aging behavior and sensitization of austenitic steels, of course we do know that we get aging of casting the steel. So it does occur in austenitic materials. And some of these high temperature materials, in fact will develop for stability a temperature. But again, it needs to be shown that the materials and the condition of interest are stable. They are not -- taking place. Producing materials that were brittle, the component.

Of course that is the aging. The sensitization we are all familiar with leaves the materials susceptible to stress corrosion cracking. And the sensitization of interest here is not necessarily from the welding. We know enough about that now. But from the actual operating temperature.

The issue came up for light water reactors back in the early '80s. And some heats and materials were more susceptible to low temperature sensitization, you know from a thermodynamics point of view, look at the stability diagrams, not supposed to happen in those temperatures. But given time, we found that you do get low temperature sensitization.

And that is much more insidious because it would affect the entire surface, not just the material at the grain boundaries necessarily.

What we found for light water reactor was that generally we took about 40 to 100 years for different heats to exhibit low temperatures sensitization. So for the light water reactor, we decided, this is really not a key issue. It happens, but not in the lifetime of the plant. So with the elevated temperatures of the gas cooled reactors, small differences in temperatures, it is like rhythm.

So, even ten degrees increase in temperature could mean a good substantial reduction in the timed desensitization. So that is an issue that needs to be looked at to determine whether the materials were sensitized, therefore, again rendering them susceptible in the environment.

We would look at materials both in the as-received and the welded conditions. Again, we would conduct mechanical and microscopic studies. We would like to quantify the time and temperature for different levels of sensitization and aging, you know to determine whether it is a reasonable thing to expect during the lifetime of the plan.

And if it is of concern, of course we would have a data base for evaluating the degree of the concern and for improving codes and standards.

Well we have talked about a number of different degradation mechanisms. And it seems to me that there is an opportunity to at least evaluate some of these things by making use of components removed from the one reactor that had 23 or so years of experience, from the AVR.

Components of interest of course would be those components where we have the operating history. We need to know the temperatures and the loading on these components. So that we could determine based on design codes and standards, how much life was used up. And then by conducting research and testing, we can determine whether those expectations were real or not.

So we could determine whether some degradation mechanisms have occurred after 23 years by just looking at the metallographic structure of the components. But beyond that, we can run mechanical tests, a fatigue test and creep test. And measure what life is remaining in this component. Therefore, we get to know what was used up and see if the corresponds to the design codes.

MEMBER SIEBER: It seems to me that Fort Saint Vrain operated at much lower temperatures than these advanced reactors.

MR. MUSCARA: Than AVR?

MEMBER SIEBER: Yes, have we learned anything from Fort Sain Vraian?

MR. MUSCARA: I am not sure about any tests that were done.

MEMBER SIEBER: From a materials standpoint?

MR. MUSCARA: One of the things we learned was that you do pick up things like chloride from the graphite itself that cause stress corrosion cracking on those components. They did experience SCC from the chloride. Of course they had problems with the water ingress and the problems with that.

But with respect to the environment, the small amounts of chloride that essentially leak gas from the graphite cause the cracking in their components.

MEMBER SIEBER: Now you would get that in a pebble bed from the graphite balls that are non-fuel balls?

MR. MUSCARA: Yes.

MEMBER SIEBER: I would presume.

MR. MUSCARA: Yeah.

MEMBER SIEBER: Okay.

MR. MUSCARA: For the issue of the in-service inspection and continuous monitoring, as I mentioned, there are long operating periods between the short duration outages. So the ISI intervals may be long. And the amount of inspections limited mostly due to accessibility problems. So we need to re-evaluate the effectiveness of different ISI programs. Taking into account both the reliability of the inspection, but also the degradation mechanisms that are possible. And taking into account those components that cannot be inspected by in-service inspection.

MEMBER SIEBER: I would think though, early on the designer along with some help from the staff would try to make as much of the plant inspectable as they could as opposed to having ISI come along as an afterthought. And you can't get into the curves and you have a lot of partials and things like that.

MR. MUSCARA: That is quite a reasonable expectation. And in fact, the ASME code requires the components to be constructed in such a way that they are accessible for inspection.

MEMBER SIEBER: But they aren't.

MR. MUSCARA: But they aren't. So they come in and ask for relief.

MEMBER SIEBER: Right.

MR. MUSCARA: And in fact when I brought this question up with the Exelon pebble bed, so it has to be realized there are some important components that can't be inspected. We plan on requesting --

MEMBER SIEBER: Relief.

MR. MUSCARA: Relief. Not at the design stage. I mean this is the time when you try to make components inspectable. You don't come in and ask for relief because we can't inspect it even before you design it.

MEMBER SIEBER: Because you don't feel like designing it. You know, inspectability is built in.

MR. MUSCARA: So it violates, already, the guidance of the code.

MEMBER SIEBER: Well I would think that

would be an important consideration up front. You know, when somebody comes in with a design concept that should be one of the rules. It ought to be inspectable. It ought to meet the code.

MR. MUSCARA: Yeah, I think from a technical point of view and policy point of view, one of the things that we could be considering is that given in-service inspection can be conducted infrequently, when components are not available, should we require continuous online monitoring. And that is one of the research areas also that we have planned.

The evaluating in-service inspection programs themselves, we would plan on conducting work using our risk-informed inspection guidelines to determine how important it is to inspect components. And for that results in an ineffective inspection, then we need to consider the continuous online monitoring.

The work on continuous online monitoring has been conducted for light water reactors. And we have developed a technique acoustic emission monitoring. For both obtaining the initiation of cracking and for following the crack severity as the plant is operating.

MEMBER SIEBER: I have a question about that particularly as applied to a gas reactor. Acoustic monitoring, listens for basically sound effects from the development of cracks in piping and so forth. For example, frequently it is used when you do hydrostatic tests as a way to determine whether you are leaking or not.

On the other hand, if I have a high speed compressor in a turbine operating, is that going to swamp out your ability to hear these things. Or can you discriminate among the sounds well enough to differentiate between the actions of the stress from the mechanical equipment that is out there running?

MR. MUSCARA: In fact, we had about a ten year research program back in the late 80s and mid 90s.

MEMBER SIEBER: I remember that.

MR. MUSCARA: -- in this area. And one of the key issues is if I have acoustic emissions is that because of cracking or some other noise source. You can't really mix the two.

So we did quite a bit of work in developing methods for discriminating noise from crack growth noise. And after many years of work, we found a very simple idea that happened to work or not even looking for this. But we conducted the laboratory work and then were ready to conduct work at an intermediate scale vessel at MPA Stuttgart. And then eventually monitored an actually plant.

Some of the work we have conducted was how do the transducers behave under the high temperature/high humidity environment. How does the coupling behave. Well we decided eventually that we needed to use a wave guide to get away from the problems of having the transducer directly on to the hot surface. So if the wave guide is coupled to the vessel or a pipe, and it is moved out of the hot area. The transducer then is coupled to the wave guide. And we conducted some tests using this technique for getting away from the temperature.

MEMBER SIEBER: The guide did the discrimination?

MR. MUSCARA: What we found was the guide did the discrimination. The sharp rise time signal from the cracks produces three mode converted sound waves. And so they are depending on the length of the wave guide, they are spaced at specific distances apart. And the white noise from other noise sources doesn't behave that way. So what we found was almost a 100% reliability, in discriminating cracking from noises, just through the mode converter sound wave in the wave guide.

MEMBER SIEBER: That is interesting. I remember the issue coming up and the problem with it because we had tried a couple simple things ourselves. But then I never followed up to find out how the problem was solved.

MR. MUSCARA: We had up to this point, we had developed euronetworks for discriminating noise from crack growth noise. And that was about 80 - 85% effective. But the wave guide was much simpler and much more effective.

MEMBER SIEBER: Cheap.

MR. MUSCARA: Cheap. So we have done this work for light water reactors and as I mentioned, with a large scale testing in and fact we monitored the Limerick reactor on a stress corrosion cracking at a nozzle. And what we found was that the acoustic emission could detect the cracking. Could characterize its growth. It could match the UT results.

Unfortunately after to one cycle, we monitored for two cycles. After one cycle the cracks stopped, you ran into a compressor stress field. And the crack stopped and the utility never removed the pipe for severe finding validation. But we did measure the crack growth and had estimated its degree of cracking.

MEMBER SIEBER: Well it would seem to me as a regulatory alternative, for example, if a licensee wanted to consider the coolant piping the same as the vessel, that this would be an acceptable alternative that you would require provided there is a good technical basis would show you that it worked. Because it doesn't sound too expensive.

MR. MUSCARA: I think the basic work has been done. It has been shown that it works in the light water reactor environment. What we would need to do with the gas cooler reactors to ensure that under the noise conditions of the --

MEMBER SIEBER: Well the spectrum is going to be different.

MR. MUSCARA: It is going to be different. And also the mechanisms. Of course, we have creep to worry about. You know, we have looked at fatigue and stress corrosion cracking for light water reactors. But of course we never looked at creep.

So there would be some additional work remaining to validate this technology for gas cooled reactors. But I think it is already a long way there.

MEMBER SIEBER: Yeah but you have a material problem just in the wave guide. Because it is on a much hotter surface than in a water reactor. I am sure you could, that one is easily solved compared with some others.

MR. MUSCARA: I think so.

MEMBER SIEBER: Well thank you, I appreciate that. That brings be closer to being up to speed.

MR. MUSCARA: Well I think to deal with the metals issues, there maybe some others, but I thought they were some of the key issues that we were considering. Moving on to the graphite.

Similarly there is a lack of data on high levels of irradiation for current graphites. There is data on the older graphites. But as we learn that the properties of graphite are very much dependent on how it was manufactured from the raw materials. Unfortunately, the raw material sources from the old graphite is gone. The mines have been closed.

And also some of the vendors. I think most of the vendors, the original vendors are gone. So the manufacturing processes in the raw materials for the new graphites would be different. Even though we striving, the industry is striving to make the graphite the same way they have done in the past. Where there is data.

MEMBER SIEBER: But the old reactors, like the N reactor, had these huge blocks of graphite with holes in them. And that to me would be a lot different than the codings on these particles or the graphite balls. Because they are discharged on a regular basis. And don't exhibit that long term distortion and growth that you would get out of a massive block of carbon.

CHAIRMAN KRESS: Yeah, but the reflector --

MR. MUSCARA: Of course I am not addressing the fuel portion. This is just the reflector, structural components --

MEMBER SIEBER: Yeah, the reflector is bigger blocks, okay. Thank you.

MR. MUSCARA: But in addition, the graphite, the pebbles do we have a graphite layer?

CHAIRMAN KRESS: They have a graphite coating.

MEMBER SIEBER: Yes they do.

MR. MUSCARA: Right. That layer also is not graphitized at the high temperatures that the rest of the graphite is. It is a much lower temperature.

And so it behaves differently than the reflector graphite.

CHAIRMAN KRESS: The matrix inside, could it be called a graphite or is it more just a carbon. I don't know if I would even call that --

MR. MUSCARA: Both graphite and carbon.

MR. CARLSON: It is sometimes called a "graphitic material."

CHAIRMAN KRESS: Graphitic material.

MR. MUSCARA: There is also a lack of predictive capability for the irradiated graphite properties from the unirradiated prosperities. Of course, I'm sure you follow the light water reactor work. For many years we have been working trying to correlate embrittlement in pressure vessel steels, and there is still work to be done there, but in the graphite we just have absolutely no work that has gone on to try and relate those properties.

In my mind that is an issue because as I mentioned, the graphite properties will vary. The irradiated properties based on the raw material properties. And the raw material properties vary as a function of the source and manufacturing process.

CHAIRMAN KRESS: Now in the case of the reflector, what are you worried about? It is not a structural --

MEMBER SIEBER: It doesn't carry any load.

MR. MUSCARA: I guess I have a couple of view graphs that will address that.

CHAIRMAN KRESS: Okay.

MEMBER SIEBER: Yeah I would think that it could just grow anyway you wanted them. All you would have to do is provide enough space.

CHAIRMAN KRESS: I would think in the prismatic concept you have a problem.

MR. MUSCARA: But the point was, that every time a new graphite comes a long, then you would need to have a comprehensive irradiation program because you know it is a little bit different, it will behave differently. And my thought is that we need to have a methodology that allows us to go from the unirradiated properties to the irradiated properties. No work that's gone on to try to relate those properties. In my mind, that's an issue because as I mentioned, the graphite properties will vary. The irradiated properties, based on the raw material properties and the raw material properties vary as a function of the source and the manufacturing process.

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reflector, what are you worried about? It's not a structural?

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MR. MUSCARA: But the point was that every time when your graphite comes along, we need to have a comprehensive irradiation program because it's a little bit different. It will behave differently.

And my thought is that we need to have a methodology that allows us to go from the unirradiated properties to the irradiated properties.

CHAIRMAN KRESS: You need a theory mechanism.

MR. MUSCARA: Mechanism and a lot of experimental --

CHAIRMAN KRESS: A lot of experimental to back it up.

MR. MUSCARA: There's also lack of oxidation, kinetics data for graphite, again, for the newer graphites.

The pebble bed reactor folks have suggested that they would use the graphite properties from the experience with the British reactors, with the sleeve reactor. Well, that's a much thinner component. It's manufactured differently. So it's not clear that the properties from the sleeve graphite in the experience pertains to the large block graphite used for the high temperature gas cool reactors.

And again, there's a lack of codes and standards for nuclear grade graphite. Very surprising for me, there's not a material specification standard for nuclear grade graphite. So we can -- the designers effectively use the information and the properties given to them by the manufacturer and they're fairly comfortable with this in that they make use of the design, that they did in the design.

My concern is, for example, if I have a graphite that is for some reason a very low tensile strength, the component is going to be thicker than it would normally be, so the designer feels he's addressed his problem. It's thicker, lower strength. We're fine. But there's some underlying reasons why this graphite is set for strength. Maybe it's successive cracking or porosity which although the component is designed thicker, those cracks will propagate during operation and cause failure of the graphite. So it's not enough to just use the properties from the particular batch of graphite. We must have certain minimum requirements for the graphite.

In addition, we need to have requirements for things like impurities which can leave the graphite and cause degradation of other components.

MEMBER SIEBER: In the reflector though, let's say the graphite cracks and you know, it's just in a can, right? And so why do you care, other than somebody else has to go and replace them.

MR. MUSCARA: Some of these components, the control rods are inside the graphite log, so that we have distortion. Then you have a problem with inserting the control rods.

MEMBER SIEBER: Right. So you make the channel bigger, right? Well, seismic is an issue if they really shift during a seismic event and so on.

MR. MUSCARA: It provides the structural integrity for the core in the core geometry.

I think we may have mentioned some of these items already, but the current data is for the old graphites. Irradiation degrades the physical thermal mechanical properties of the graphite. These changes can cause stresses during operation and loss of integrity.

The strength of graphite initially increases with irradiation dose and then at higher level it begins to decrease.

The dimensional changes that initially graphite begins to shrink and then with increasing radiation it begins to swell. And then, of course, beyond the turn around, the graphite loses an entire structural integrity. It essentially falls apart.

As we mentioned, the loss the structural integrity, the loss of core geometry and potential problems with insertion of control rods. So we would need to study the changes that undergo in the graphite as a function of the levels of radiation and temperature.

I guess with respect to temperature, I want to mention that if we irradiate these materials at higher temperature, that's not necessarily a conservative direction to go into. For example, we discussed a little this morning getting margined by doing higher temperature exposures of the fuel. At higher temperatures, you get some annealing, so going up to a higher temperature to study radiation effects is not necessarily a prudent thing to do. I mean we need to go there if we experience those temperatures, but irradiation at lower temperature sometimes can be more detrimental because it does not anneal out the damaging effect from the irradiation.

MEMBER SIEBER: I thought in decades ago that was how they would run a graphite reactor at very high temperature for a while to try to recover the graphite physical properties and basic dimension.

MR. MUSCARA: You can anneal out some of the irradiation and also having a little creep, it helps at the beginning that you are relieving the stresses. Of course, you're getting too much creep with the material starts to flow. It's not a good thing.

MEMBER SIEBER: On the other hand, the net effect of that is to make it more brittle and less weaker?

MR. MUSCARA: With the irradiation?

MEMBER SIEBER: With the annealing? Or multiple annealings?

MR. MUSCARA: On the graphite?

MEMBER SIEBER: Yes.

(Pause.)

MR. MUSCARA: Anyway, as far as research

in the area, we would intend on reviewing available high dose irradiation data. Some data has been taken, for example, at Oak Ridge, under a DOE program. That data was never analyzed because they ran out of funds. We would hope they would have access to the data to analyze it.

We would conduct irradiation tests on test reactors, high flux test reactors, different temperatures, different irradiation exposures. And we would conduct microstructural evaluations, dimensional, mechanical, thermal and physical property measurements, both before and after the irradiation.

As mentioned earlier, this kind of work is very, very expensive and clearly it would also be depending on international cooperation to get some of this information.

Again, I brought up the issue the need to have correlations between the unirradiated and the irradiated properties. These properties depend strongly on the raw materials and the manufacturing process. Some data is available from old graphites, but no data on the new graphites.

In the conducting research, what I would hope is that we could more or less piggyback on some other work that's going on. I can get to this a little bit later, but the European Community is already planning on conducting some extensive irradiation testing of five current day graphites and I would hope that we could conduct some parametric studies along with those studies to evaluate some of the changes in the raw material properties and how this affects the irradiation.

So there's work that's going on, but the work could be augmented to try and get at not only, for this particular graphite, this is how it responds, but trying to get some correlations for the important parameters to predict how those parameters affect the irradiation behavior.

(Pause.)

Again, this will be the kinds of studies we would conduct. I think I've mentioned most of these already. Temperature irradiation levels, raw materials makes a big difference. And processing parameters, they manifest themselves into the properties of the as-received graphites. There are many different ways of getting to the same properties. So just looking at processing parameters might not give us the final result, but we need to keep in mind when we develop a matrix of tests what are the important processing parameters that affect the raw material properties. And they make sure that those things are incorporated.

There are a lot of different parameters, both processing and initial properties that need to be looked at and we need to do a careful job of selecting and evaluating which parameters to use for studies. To do this, my thought was we get together a group of experts and discuss what are the potential most important properties that might affect the irradiation.

So there would be several workshops held before one would even develop a test matrix for this kind of work.

In addition, I'll mention it later also, but we have acquired a graphite expert for our branch who will be working in this area and he has an assignment, about a 3-month assignment in the U.K. to take advantage of the experience and knowledge that's been gained there and also make use of the experts that are available to start developing some of these test matrices.

CHAIRMAN KRESS: Do you think three months is enough time for him to --

MR. MUSCARA: Probably not, but at this point I thought that's what we could afford. It would be a good first try.

CHAIRMAN KRESS: When you run these tests on graphite, irradiating and see what effect it has on the properties, do you need large specimens or can you do this with small?

MR. MUSCARA: That is an important question. That's something that needs to be decided. My view is that the property will change through the thickness -- the raw material properties, therefore the irradiation properties. And we need to know what those properties are as a function of thickness. So I think we need to be very careful about what select and as a minimum have samples from the surface and some intermediate locations going through the center of the component.

MEMBER SIEBER: Well, the fluence varies through the ball section --

MR. MUSCARA: Sure.

MEMBER SIEBER: So the properties will vary at a right angle.

MR. MUSCARA: The irradiated properties. But even the raw material properties. The chemistry will change through the thickness, the density, the porosity --

CHAIRMAN KRESS: Yes, that will wear more than fluence because you can take care of the fluence otherwise.

MEMBER SIEBER: So you can calculate it out.

MR. MUSCARA: Right.

MEMBER SIEBER: Would the ultimate outcome of this kind of work result in a standard? I would think that would be a good way to codify how you're going to use it and what pro than fluence because you can take care of the fluence otherwise.

MEMBER SIEBER: So you can calculate it out.

MR. MUSCARA: Right.

MEMBER SIEBER: Would the ultimate outcome of this kind of work result in a standard? I would think that would be a good way to codify how you're going to use it and what properties it ought to have. Or would you have it as a reg. guide or --

MR. MUSCARA: Yeah, I think the effects of irradiation, how it affects the properties, needs to become part of a standard, a design standard.

MEMBER SIEBER: Right. I agree. Well, I was thinking in terms of a national standard like ANS or somebody like that.

MR. MUSCARA: Well, oxidation kinetics so it's another area where there's a lack of data. This information is needed for evaluating the heat generation, the structural integrity, and core geometry during normal operating and accident conditions. The air ingress, of course, would lead to corrosion and oxidation of graphite. It's an exothermic reaction so we need to know how much heat is generated and of particular importance during an accident condition.

There's a loss of material surface, so the structural integrity could be impaired. There's a reduction of fractured toughness and in strength of the graphite with the oxidation and changes in thermal conductivity. So all of these parameters are important for safety review and evaluation.

CHAIRMAN KRESS: Now are you interested in the cases where you have an air ingression accident that could lead to rapid combustion or rapid -- or are you interested in low levels of contamination of oxygen and helium? This long term degradation effect.

MR. MUSCARA: We're considering both. So one of the bullets here has to do with the amount of oxidant in the atmosphere. So for as low air ingress it would be one level; for break would have much more oxidants available to oxidize the graphite.

We're interested also in different kinds of graphites. The graphite, you say the pebble graphite which has not been graphitized at high temperature will have a different rate of oxidation. We're interested in evaluating the oxidation rate of graphite dust. The dust will deposit on surfaces, but if it's, you know, we have an accident now, it's the graphite dust in a given surface, it oxidizes faster.

Heat generation. We need to know how it affects the particular component.

CHAIRMAN KRESS: Now when people do these kinds of tests usually they do them with small specimens. Now the questions comes up on the effects of an air ingression accident. Will the graphite itself burn or have a sustained oxidation process? And that generally is a geometry problem and how much heat are you generating and how can it dissipate in various directions and how much oxygen you can get there to produce the combustion.

Do you have plans for some sort of look at that question, the combustion of large chunks of graphite?

MR. MUSCARA: It's a question, but I don't think we've defined how to go about conducting those tests.

CHAIRMAN KRESS: But that's not what you're talking about here. This is something else.

MR. MUSCARA: I think it's both. I mean we need to know from the dust to the large component, how that affects the rates in carrying away the heat if it's a large component. So we would measure the heat generation and the oxidation rates, both.

Well, we talked about the variability of large block graphite and want to use information from thin section graphite. Again, the designers, because of lack of data, where hoping they could use data from the sleeve graphite, but that's a lot thinner and it's not clear that's applicable. So we need to conduct more in this area to determine the differences in the graphite through the thickness, both in properties, chemistry, things like porosity, distribution, and numbers.

We're not planning on irradiation work as a function of this variation in block thickness, but if we evaluate the changes in properties in the raw graphite, and if those changes are considerable, we have to be able to estimate whether irradiated properties would respond also to a large degree.

CHAIRMAN KRESS: What I envisioned earlier when I thought to ask you had to use big specimens to do the testing. I thought maybe you could use the big specimens that were sectioned right and look at their property variations and put each of those sections in the same fluence area and that should test a lot of small specimens representing one big one.

MR. MUSCARA: Again, we're going to take advantage of work going on in Europe and Japan in this area. They are planning work both in oxidation and irradiation. I'm not sure how the tests are really being set up, but that's why I suggest we have an expert group meeting to define those tests.

CHAIRMAN KRESS: That's probably a good way.

MR. MUSCARA: Well, the lack of codes and standards in nuclear grade graphite, again, I think most of these things I've mentioned with respect to the issues, but there is a lack of design codes for taking to a concrete fatigue strength fracture toughness. We need material specification that established the minimum requirements for mechanical, physical, and chemical. We would need to limit elements that may be detrimental to the irradiation properties, or elements that can cause degradation of other materials. For example, the chloride that we had experience with.

With respect to the specification, I've contacted ASTM staff to discuss whether there's a potential for them to develop in nuclear grade specification for graphite. And they agreed that they should and can develop such a standard and their activities are already in place to develop a nuclear grade graphite specification. We're supporting a little work on that at Oak Ridge National Laboratory.

We're providing one staff person who's an expert in graphite participating the code committee, or the ASTM committee. He's also, I guess, the chair of one of these committees. So we're providing a little bit of support, and also our staff is participating with that specification development.

In the area of design codes, there is very little information. There's no national codes for this. The U.K. and the Japanese have developed some aspects of design codes in these areas. We would hope to be able to get some information under the cooperation on their design process. But the initial parts of the research will be to review and evaluate what's already available from these two countries and see what improvements need to be made and then work with codes and standards committees to develop the design codes.

CHAIRMAN KRESS: Dana Powers had an issue with this graphite, it has something to do with energy build up through the irradiation. It's different than the Wigner energy, but it has higher level components to it that don't get annealed out to operating temperature. And he maintains that these could have significant energy releases during an accident condition when you get up to the higher temperatures and suddenly release these things. Does your research plan to look at any of that or the different --

MR. MUSCARA: It was not discussed in this current plan.

MR. FLACK: Yeah, we do recognize that and the plan. I think Don brought it up, Carlson.

MR. MUSCARA: As I said, it wasn't discussed in the materials.

MR. FLACK: If it wasn't in the materials part, I guess is the issue. I don't see Don here. Maybe you can bring it up.

CHAIRMAN KRESS: It's probably a severe accident issue or something.

MR. FLACK: Yeah, at the high temperatures the effects -- it seemed, the indication seemed, oh, Don just came in. Don Carlson will be up in a little while to talk about the nuclear analysis part of the plan.

The question had come up on graphite's behavior at higher temperature and not the Wigner energy, but the energies of releasing graphite at higher temperatures. The part, I believe there's a discussion of part of that in the plan.

MR. CARLSON: Yeah, I mention it in the nuclear analysis part, not that it's really a nuclear phenomenon but something you have to add to the decay heat power in analyzing these events.

CHAIRMAN KRESS: So it's in there?

MR. FLACK: It's in there. Yeah.

CHAIRMAN KRESS: Dana will ask that.

MR. FLACK: I'm sure he will. That's why we added it.

(Laughter.)

MR. MUSCARA: So I mentioned working with ASTM, eventually probably will work ASNE once we get some information about U.K. and Japan has been using. And as I mentioned earlier, we'll have a staff assignee to work in the U.K. to start addressing some of these issues and develop some consensus on what are the important parameters. For example, for the material specification, what are the important parameters for inducing irradiation damage.

As I mentioned, we do plan on establishing some international cooperation in the materials area, in particular, with Japan and with the European Communities. We have visited a number of countries to discuss materials issues among other issues. And we have shared our thoughts about research needed. Pretty much the thoughts are in the plan with both Japan and with the European Communities.

We've met the European Community representatives. They've reviewed our plan in one of their own independent meetings in Brussels. In effect, in the materials area, they decided that all this was important work and work that should be done. Some of the work is ongoing in their current program, but much of the work will be picked up in their next HTRM, M standing for materials program.

That's their sixth technology program. It will initiate in 2003. They're putting out requests for proposals at this time. They expect proposals at the end of this calendar year and they will initiate funding of their sixth program as I said in 2003.

So we have discussed participation with the EC and Japan and we're in the process of developing a draft agreement to do this. There is no exchange of funds, but it would be an exchange of research results from each other's programs. Some of the work going on in the European Communities, they're looking at a pressure vessel material for the high temperature gas cool reactor, but probably the next generation they're looking 9 percent chrome material. Of course, Exelon was planning on using the standard light water reactor material.

I believe at one time GA was intending on using the two and a quarter chrome 1 molley, but I think now they're also considering the 9 percent chrome. So the Europeans have begun work on the 9 chrome material. They will irradiate the material, both in welded and unwelded conditions. And they'll be conducting fatigue creep tests and irradiation tests.

They're also looking at two turbine blade materials. One material is aluminum, the other is chromium. So they have a chromoxide or an alumoxide coating that would form as a protective coating. And they're trying to determine which one might work best in a heating environment.

There's some work that they were planning on doing in the new program on in-service inspection methods, not necessarily evaluating the efficiency or the effectiveness of these inspections, but different methods that are needed for inspecting the reactors. And they also have begun some work on irritating graphite. As I've mentioned, they have five different graphites that they're going to be studying.

CHAIRMAN KRESS: Was there any work done in Canada with graphite?

MR. MUSCARA: Actually I don't know. I haven't looked.

CHAIRMAN KRESS: Mario seemed to think there was.

MEMBER BONACA: I thought they did some work in 1998.

MR. MUSCARA: We have looked at some of the literature. I'm sure not exhaustive, but nothing popped up from Canada. Most of the work I've seen has been European Communities and Japan. Of course, a lot of work is going on in Russia.

Well, some of the research that may not be picked up is international programs, at least not to the levels that I would like to see. It's work on the effect of the impurities on the degradation of materials. On the effectiveness of the service inspection is using a risk informed method for evaluating their effectiveness and on the correlations for the non-irradiated properties to the irradiated properties. And I believe that exchange of research results in these areas will buy us the results from all the other work that has been planned by the European Community and by Japan.

In addition in Japan, there has been considerable work done on the design and also on high temperature corrosion. And hopefully, we'll get access to that work also.

CHAIRMAN KRESS: There was some attempts to make correlations on non-irradiated material, properties for metals and had to do with dislocations and effects on the matrix. Is any of that applicable for graphite or completely different?

MR. MUSCARA: I am not sure. I have discussed with several experts. I think how many people talked what they said I would never get any correlations. Too difficult. Too many parameters.

CHAIRMAN KRESS: That's what I was wondering.

MR. MUSCARA: Others are fairly confident that now we could develop some correlations.

CHAIRMAN KRESS: It's certainly worth to look at it.

MR. MUSCARA: I've asked, I said we split about 50-50. I know it's been a lot of extensive work done in just the pressure vessel steel.

CHAIRMAN KRESS: Well, you know it doesn't look like trying to make such a correlation would be all that expensive because you're going to get the data anyway.

MR. MUSCARA: You certainly would get it, let's say, for one heat. But what we want to do now is for similar heat vary some of the important parameters. And then you have to be exposed to the irradiation. And you conduct side by side tests. And this is one thing that I suggested to the European Community that they're doing the other extensive work on five graphites, we ought to get together and decide on how best to make use on that work by doing some parametric studies on the side but coordinated with what they're doing. They liked the idea. They'd like to pursue that. But you can say the camp is divided at this point whether we're going to be successful in developing these correlations.

CHAIRMAN KRESS: That's always the case.

MR. MUSCARA: And I think if you look at the pressure vessel steel, you know, maybe they're right. This is much more complex material than steel.

MEMBER FORD: Joe, coming back to the whole question of privatization which we have based on something presuming to do with the risk. Half your input to that decision making process will come from other organizations. Don't necessarily have the same drivers as you will. So how useful is this specific data that's coming from the European Community or Japan? How useful will that be to solving your particular prioritized target? Do you understand what I'm getting at? You've got no control over what they're going to do. They may be completely irrelevant.

MR. MUSCARA: We have some especially for the new program. In fact, when I sent them our program, and they reviewed and I went back and discussed it, they said this is great. This is exactly what we need to do. Go do it.

MEMBER FORD: You said your program. What was in this document, the red one?

MR. MUSCARA: Yes.

MEMBER FORD: Okay.

MR. MUSCARA: But they were not as excited about some of the areas that I mentioned. So maybe they will take a little of the area but not as much. And the idea was there was that we would exchange information.

MEMBER FORD: When they say great, that's exactly what we need, is that because they weren't doing it?

MR. MUSCARA: They pretty much started out doing some literature reviews and assembling some data bases. They had done this for graphite, for pressure vessel material, and for turbine based material. Now that they've done that, now they're going beyond it. Now they need to get into doing research.

MEMBER FORD: So they haven't done any data collection themselves?

MR. MUSCARA: Very little so far. They've just initiated a pressure vessel program and their graphite, they purchased the graphites that they're going to expose. So, you know, they started about four years ago but a lot of it has been coming up to speed. What has been available? Where do they want to go? And what needs to be done? And so our plan came in about the right time, I think.

MEMBER FORD: That applies to both the United Kingdom as well as --

MR. MUSCARA: Well, this was more the European Communities. I'm not sure what role the United Kingdom is playing in this HDRM program. They have had, of course, on the graphite area lots of wrong data and experience. But as far as how does it apply, when we're working and reviewing the PDMR, and I looked at what Europeans were doing, my first thought was well, this is great, but it doesn't help me right now. Because they're looking at the next generation of steam generators. They're looking at higher temperatures. For example, the 9 chrome material.

So at one point I thought this is not

going to be that beneficial to us. But as the General Atomics design comes along and PDMR sort of is on the back burner for awhile, that work seems more and more appropriate. Because we were thinking ahead as to what might the materials be for the next generation of high temperature --

MEMBER FORD: And this international society of takeovers, etcetera, are any of the OEMs in Japan and European Community involved in this work and therefore by inference maybe General Atomics?

MR. MUSCARA: I don't think I understood.

MEMBER FORD: In collecting this data for the European Community HTR project and for the Japanese JAERI program, are any of the commercial manufacturers involved in this work?

MR. MUSCARA: Yes, some of the European work. In fact, the research will probably be conducted by people, for example, in the blade material. Some of the companies producing the material will be doing some of the research. So within the European program, it's not necessarily our national laboratories. A lot of commercial groups doing the work. In Japan, a lot of it is JAERI.

MEMBER FORD: Okay.

(Pause.)

MR. MUSCARA: I think, I guess, I'm at the summary. Did I pick up some time?

Well, we discussed a number of key technical issues and this relates to the chromes and standards of the availability and applicability of these standards. The lack of data in correlations for graphite. In my mind, environmental effects and degradation materials are a very important area that is not very well addressed. The pipe as a vessel, again, it's for the technical and the policy issue.

We need to determine whether that can be treated as a vessel based on the experience we've had and the lack of the experience for these materials to be used in a gas coal reactor.

CHAIRMAN KRESS: How does it compare in thickness to the vessel?

MR. MUSCARA: Typically the thickness of the duct pipe is about 1.6, 1.7 inches thick. So it's very much --

CHAIRMAN KRESS: Probably looks like a pipe.

MR. MUSCARA: It's a pipe. I asked this question in China about the pipe on the break and they essentially said to me no, we considered our vessels so we could avoid doing an analysis.

(Laughter.)

They could not do a smaller design for it.

CHAIRMAN KRESS: That's a wrong way to make a decision.

MR. MUSCARA: And I think that's a trick that has been played. It's not necessary because it really believes and behaves like a vessel. I think it's just get around this environment.

CHAIRMAN KRESS: If you had a risk basis for saying that this thing is not going to break at a certain frequency, then maybe you can do something like that.

MR. MUSCARA: And at that this stage I don't see how they can make the case without the data on the environmental effects, for example, and the appropriateness of creep and fatigue in their interaction.

CHAIRMAN KRESS: I don't either. That's the most likely place for a break.

MEMBER ROSEN: Your case wasn't made in a light water reactor with about 3,000 reactor years of experience in the United States. The case is now being attempted to be made based on experience that the largest pipe in the pressurized water reactor won't fail in a double ended guillotine manner. And there seems to be some staff acceptance of that, that it's going to be a very low probability event. But there's 3,000 reactor years of experience at the relevant conditions.

Now, to say the same thing is true for a plant without any experience just --

MR. MUSCARA: In different conditions, in different temperatures.

MEMBER ROSEN: At much higher temperatures.

MR. MUSCARA: It's a slight stretch.

MEMBER ROSEN: It's a big stretch.

CHAIRMAN KRESS: I think it's a stretch of misapplication on the design basis concept, too. Because in my mind the design basis concept says you select design basis accidents and you prescribe how you analyze them in a conservative way with certain tools and you have selected theories of merit for acceptance of the design. And you do that and lo and behold the whole reactor turns out to be safe over the whole spectrum of accidents.

Now the reason is because when you put in provisions and do the defense-in-depth parts that are required in the design basis case, those also deal to some extent with severe accidents.

MEMBER ROSEN: Most of them.

CHAIRMAN KRESS: Some of them. Most of them. So to take away one because oh, this isn't severe accident space because its frequency is so low it's not going to happen, is not the right concept design basis accident. You have to ask yourself if I take that away, have I now done something to the reactor that would put it a such a higher risk level that it's an unacceptable risk?. And I think that's kind of missing from that concept.

MR. MUSCARA: Even the data we experience we have today is especially for stress corrosion, cracking, and piping, and nozzles. We may not have had a break, but I think some cases might have come close. I mean, Duanne Arnold for example. Talk about this pipe. This thing have been of concern to me with respect to degradation.

I mentioned earlier that one of the designs there are jackets of insulation. They are going to the inside of this pipe.

Well, these jackets are about a foot to two long. And so they're several of these pieces that go in, which means I'm now naturally creating crevices. And has anybody looked at crevice corrosion cracking with the environment of the pure helium? And talking about that pipe being treated as a vessel. I mean, I can almost see a mechanism right now that could occur in these pipes when you have the insulation on the inside and creating crevices.

MEMBER ROSEN: Well, Joe, I don't think you need a lot more encouragement from the Committee to hold you position. I think you heard at least from myself and Tom and few others.

CHAIRMAN KRESS: Is that pipe still concentric? In the GTMHR concept it used to be a concentric pipe with a hot guise going one direction and the cold guise going back the other way. Is that still?

MR. FLACK: I believe it's the same design.

MR. MUSCARA: Let me sort of finish with the summary in just a few more words. So we haven't taken this lightly. We've looked at potential issues. We've written about them, discussed them. We in fact have initiated two small projects. One at Argonne National Laboratory to look at the basis for the design codes and standards for metals and to review in more detail than I have what information is out there on the effects of impurities, because I think that's a key area. And at ORNL we've started a project to start working with the standard specification, also to review what data and information on the potential for developing the correlations from the unirradiated to the irradiated properties. We planned on a having a 3-month assignment in the U.K. so we can learn more about graphite technology and experience and Dr. Srinivasan who was on our staff will be taking on the assignment.

MEMBER WALLIS: Do you have any problem with the language?

(Laughter.)

MR. MUSCARA: Really? Do we have any problem with the language. That's it.

CHAIRMAN KRESS: Thank you. I'd like to get a feel from the Committee whether they need a break or not.

MEMBER SIEBER: Sure do.

CHAIRMAN KRESS: This looks like a good time to take a 15-minute break. Why don't we come back at 25 after. 3:25.

(Whereupon, the foregoing matter went off the record at 3:12 p.m. and went back on the record at 3:27 p.m.)

CHAIRMAN KRESS: I think we'll get started again. We have most of us here.

MR. FLACK: Okay. Don Carlson and Richard Lee will now present their part of the plan, which deals with the reactor plant analysis.

CHAIRMAN KRESS: It's always a pleasure to have Richard here. We have him here so seldom.

MR. CARLSON: Okay. Again, my name is Don Carlson. I'll be presenting this with Richard Lee. It's about reactor systems analysis for advanced reactors.

The scope of reactor systems analysis encompasses three technical disciplines: nuclear analysis, thermal-hydraulics analysis and severe accident and source term. The research program will provide some data and validated system analysis tools that are appropriate for predicting system conditions and system responses in advanced reactors. A key point that you may have noted from Joe Muscara's talk is that, for example, the irradiation properties of graphite change such that thermal conductivity goes down considerably with irradiation if it is a function of irradiation temperature.

And a unique aspect of the new HTGR designs is that the maximum fuel temperature reached in say a conduction cooldown event is very strongly dependent on graphite thermal conductivity. So this hopefully puts some of those issues into a useful perspective.

CHAIRMAN KRESS: So your thermal-hydraulic analyses have to use the most irradiated, worst degraded properties of the graphite or --

MR. CARLSON: Exactly. For example, if you were doing a test in a prototype facility, if you did that early in life, you would get lower maximum fuel temperatures than if you did it toward the end of the graphite life.

CHAIRMAN KRESS: This is the concept of licensing by test?

MR. CARLSON: Yes, yes.

CHAIRMAN KRESS: It would have to have -- okay. There's some issues there.

MR. CARLSON: So these systems analysis tools that we'll be providing will allow the staff to independently check or confirm the applicant's analyses and get a better understanding of the technical issues, uncertainties and safety margins. The systems analysis will then also contribute to developing the regulatory framework by assisting in the identification of safety significant systems, components and licensing basis events.

The research plan addresses the three

major disciplines in separate subsections. I wrote all the sections in the plan dealing with nuclear analysis, both for reactor systems analysis and for the other three regulatory arenas: materials safety, waste safety, and as I mentioned earlier, we have a placeholder for safeguards as well. And all of those areas are heavy on nuclear analysis. But today we're talking only about nuclear analysis for reactor safety, and I'll be presenting that.

Richard Lee will be presenting the parts about thermal-hydraulics analysis and severe accident and source term analysis. That was the work of several different co-authors: Steve Bajourck, Tony Ullses, a little bit from me on HTGR thermal-hydraulics. Steve Bajourck was advanced light water reactors. Steve Arndt also wrote some of those input. And in the severe accidents area, Chester Gingrich and Ali Bebihani contributed those parts of the plan.

Now moving into the nuclear analysis area. Nuclear analysis is perhaps a term that has not been widely used in the NRC. I'm not the first to use it, but it encompasses everything concerning the interaction of radiation with matter. That is how I define the technical discipline. And so in the area of reactor safety, it would encompass core neutronics, both static and dynamic, which would include the evaluation of reactivity transients, temperature feedback coefficients for the fuel moderator and the reflector, reactivity control and safe shutdown and also would deal with spatial power distribution issues and issues such as local power peaking and oscillation stability.

Another type of calculation that's done in nuclear analysis is nuclide generation and depletion, sometimes referred to as nuclear transportation calculations. They're done for neutronics; that is, you analyze the core burnup to get the compositions used in your core neutronics calculations. Another main use for nuclide generation and depletion is calculating the decay heat power and also radiation sources and releasable inventories of fission products in the fuel.

A third area of nuclear analysis is radiation transport and attenuation. That would be find application for material activation and fluence damage in each TGR, as you're talking about fluence damage to graphite in addition to metallic components. And also you do, of course, radiation shielding calculations for radiation protection.

And then finally, although this isn't the subject of plan, there are nuclear analysis issues in out-of-reactor at the front end of the fuel cycle for criticality safety in the back end and fuel cycle criticality safety with burnup credit, decay heat and spent fuel, radiation shielding of spent fuel and non-destructive assay for safeguards.

CHAIRMAN KRESS: Your nuclide generation and depletion, is that origin we're talking about?

MR. CARLSON: That would typically be origin or cinder, yes.

CHAIRMAN KRESS: Oh, cinder, that's right, I forgot.

MR. CARLSON: We use origin. So starting off with advanced light water reactors, there are no significant new issues for AP1000, it's a lot like AP600 in current generation light water reactors, so the issues are the same. For IRIS, there are some new nuclear analysis issues concerning fuel depletion, modeling and validation for the fuel with five to eight percent initial enrichment that they'll be using in IRIS. The assembly lattices have a greatly increased ratio of moderator to fuel; that is, they're taking, essentially, a pin from 17 by 17 lattice and putting it in a 15 by 15 lattice, leaving more room for moderator.

They're using very strong, advanced burnable poison designs and burnup levels up to 80 gigawatt days per ton. The maximum burnups we see in current generation light water reactors are 60 gigawatt days per ton on an assembly basis.

Related to these depletion issues, there would be global core neutronics issues for the five- to eight-year straight-burn core. The IRIS does not do fuel shuffling. You load the fuel and burn it for five to eight years and then reload the whole core. The neutronics uncertainties and modeling issues would tend to compound more than you do with fuel shuffling, where in current generation reactors you have a relatively fresh assembly in close proximity to the higher burnup assembly, so that tends to wash out the effects of depletion uncertainties.

And, finally, you have decay heat power modeling and validation issues. Probably you need for an extension of the ANS 5.1 decay heat guidance that would be applicable to this new fuel and the higher burnups in particular.

Now, for some of the research activities that we would be doing for IRIS, first of all, we would identify relevant reactor physics to benchmark data. There have been light water reactor benchmark data for higher burnup fuels developed in various places in recent years. There was the REBUS Program that the NRC was involved in in Belgium. In Switzerland, there is the ongoing LWR PROTIS Program. And there were the series programs in U.K., France and the U.S. involving experiments at Catarash on the Ecole and Minerva facilities. And then there is also an ongoing nary-funded program at Sandia for doing measurements related to burnup credit that would have some applicability to IRIS.

CHAIRMAN KRESS: Now, what is this data about? Is it about the buildup of nuclides or is it about decay heat or --

MR. CARLSON: This would be critical benchmark data --

CHAIRMAN KRESS: Critical data.

MR. CARLSON: Critical benchmark data for the fresh material and for fairly high burnup material.

CHAIRMAN KRESS: Okay.

MR. CARLSON: And there would be some radioisotope assay data afterwards, destructive assay.

CHAIRMAN KRESS: But this involves all your cross-sections and --

MR. CARLSON: So there would be origin-type depletion validation.

CHAIRMAN KRESS: This involves all your cross-sections then and the --

MR. CARLSON: Yes. It involves the cross-sections and all the tools that use the cross-sections.

CHAIRMAN KRESS: And those are things like PDQ? What code do you use in these things for that?

MR. CARLSON: Well, the NRC is in the process of developing for the first time a lattice physics tool.

CHAIRMAN KRESS: PARCS, was that the name of it?

MR. CARLSON: PARCS is our diffusion theory code. It's a global 3D kinetics diffusion theory code. And we're developing a lattice physics tool that would produce data for use by the diffusion theory code.

CHAIRMAN KRESS: This is a code to benchmark against this data.

MR. CARLSON: And so those suites of codes would be benchmarked against these data.

CHAIRMAN KRESS: Putting in the right cross-sections and stuff for the -- well, this is IRIS, I guess it doesn't need any -- doesn't need much changing.

MR. CARLSON: Yes. Like I said, the changes are there's a greater use of burnable poisons, there's an increased moderator fuel ratio, and so we would have to look for data that gets you more into those physics regimes.

CHAIRMAN KRESS: What does that effect, the energy distribution of neutrons?

MR. CARLSON: Yes. You get a softer thermal spectrum.

CHAIRMAN KRESS: Softer thermal spectrum.

MR. CARLSON: And, of course, we're pursuing international cooperation through the AIEA, the European Commission and OECD/NEA. And these would be conduits for getting to some of these data that I mentioned.

The general approach that we would like to pursue in the international cooperation would be to use high order methods like continuous energy Monte Carlo as a code-to-code benchmark against the more proximate practical methods that you use for reactor physics.

The HTGRs, the GT-MHR and PBMR, share some similar features with regard to nuclear analysis. They both, of course, use fission products retaining coated fuel particles, graphite as the moderator and neutronically inert helium as the coolant. Moderation by graphite gives you a prompt neutron lifetime, about 20 times what you get in light water reactors. The migration links in graphite is 62 centimeters versus 5.8 centimeters in water. It takes about 114 collisions to thermalize a neutron with graphite versus 18 collisions on the average with water. So they're a very significant physics from what we're used to in light water reactors. The large migration area bottom line there is that an HTGR is much more tightly coupled neutronically than a light water reactor of similar dimensions.

CHAIRMAN KRESS: It sounds to me like those were good things you were saying about the --

MR. CARLSON: Oh, yes. They're good things.

MEMBER SIEBER: Except for the prompt neutron.

MR. CARLSON: Well, the prompt neutron lifetime is good too. It's a longer -- you get much wider prompt pulses if you get any.

MEMBER SIEBER: Okay.

CHAIRMAN KRESS: What's the issue with the long annular cord geometry? Does that --

MEMBER ROSEN: Axial stability?

CHAIRMAN KRESS: Does that cut down on your --

MR. CARLSON: At some point, you get into axial stability issues, the mode separation of the fundamental from the higher harmonics goes away eventually if you get long enough.

MEMBER WALLIS: Does the helium produce significant moderation or is that negligible?

MR. CARLSON: That's negligible.

MEMBER WALLIS: Negligible.

MR. CARLSON: Both reactors use control and shutdown absorbers located in the graphite reflector regions.

CHAIRMAN KRESS: I understand that you have significant moisture ingress, that you might have some neutron effects with the coolant if you had a leak, had a moisture leak or something, you might have a problem with?

MR. CARLSON: Well, in the old designs that use steam cycle, that was a more likely event, where you had high pressure water systems interfacing with the primary system. In these Braten cycle systems, you only have low pressure water, but still you would have to consider moisture ingress for depressurized or underpressurized conditions in the primary. And what happens in a moisture ingress is you're adding hydrogenous moderator to an undermoderated system, so K-effective goes up.

CHAIRMAN KRESS: You're adding positive reactivity.

MR. CARLSON: You're reducing the prompt lifetime, you're decreasing the migration links so fewer neutrons are getting to the absorbers and the reflectors, so you're reducing the reflector absorber work.

CHAIRMAN KRESS: Don't you have to have a lot of water to do that? I mean it's going to be steam when it gets in there.

MR. CARLSON: A little water goes a long way for slowing down the neutrons. It really takes over the slowing down term just a little bit.

CHAIRMAN KRESS: I would have thought you had so much graphite in there, you wouldn't even know if this water was there.

MEMBER ROSEN: Can you quantify that? That's an interesting result. I mean just how much is a little?

MR. CARLSON: I can't quantify that. I could, but I'm not --

MEMBER ROSEN: Well, a little helium goes a long way too but not quite as far, I guess.

MR. CARLSON: Well, helium is neutronically inert. Graphite is a very powerful scatterer, a very powerful slower down.

MEMBER WALLIS: Well, helium is a slower down too. Helium is a slower down.

MR. CARLSON: Yes, but there's just not enough helium atoms to have a significant moderation effect. It's a gas.

MEMBER WALLIS: But it's under pressure.

MR. CARLSON: Yes. So unlike --

MEMBER WALLIS: Water is going to be liquid in this thing?

MR. CARLSON: No, there will be steam. It will be --

MEMBER WALLIS: It would have to be gas too.

MR. CARLSON: The steam, yes. Helium also has a very small cross-section of hydrogen.

Unlike the earlier HTGRs, the Fort Saint Vraian and the THTR, these newer designs use thorian instead of -- the older designs use thorian and HEU; the newer designs use low-enriched uranium. In the case of the PBMR, eight percent in the equilibrium core. They start out with four percent in the initial core. And in the case of GT-MHR, 19.9 percent initial enrichment. As we said before, they have long annular core geometries with control and shutdown absorbers in the reflectors. These similarities then do lead to fairly similar modeling and validation issues for the two design concepts.

Some of the issues that are discussed in the plan, the temperature coefficients of the reactivity. It is claimed that both designs have a very strong negative temperature feedback. The components are temperature coefficient of the fuel, the moderator and the reflector. The first two are strongly negatives, and the last one is positive.

CHAIRMAN KRESS: And in fact that's the reason the temperature never gets above the 1600 because of the temperature coefficient?

MR. CARLSON: It sets itself down.

CHAIRMAN KRESS: So it's important to know that.

MR. CARLSON: In fact, one way -- the favored way of shutting these down is to simply turn off the coolant.

CHAIRMAN KRESS: It shuts it down and then you get the xenon buildup to keep it down. But the xenon decay --

MEMBER ROSEN: And all of a sudden you return to power.

CHAIRMAN KRESS: Yes, the xenon decay would come back to power then?

MR. CARLSON: Yes. After about a day, xenon decay and then you didn't put in absorbers, then you would eventually --

CHAIRMAN KRESS: Then it would just sit there an oscillate.

MR. CARLSON: Then you oscillate at low power.

CHAIRMAN KRESS: Low power.

MEMBER ROSEN: So would you say that again? The fuel and the moderator are strongly negative.

MR. CARLSON: But the reflector temperature coefficient is positive. So if we could figure out a sequence where you heat the reflector without heating the fuel in the moderator, you would have positive feedback.

CHAIRMAN KRESS: Overall, you have positive coefficient.

MEMBER ROSEN: Overall?

MR. CARLSON: Overall, you have a strongly negative.

CHAIRMAN KRESS: Strongly negative. You have a strong negative overall.

MEMBER ROSEN: Right. Because if it was overall positive, you might as well stop.

CHAIRMAN KRESS: Yes, yes.

(Laughter.)

MR. CARLSON: Well, one question that I was kicking around is when you return to criticality, if you don't scram after xenon decay, you have a combination of xenon decay and perhaps some cooling from the conduction, and you're cooling from the outside in. The peak temperatures are in the middle.

CHAIRMAN KRESS: Oh.

MR. CARLSON: And so the reactivity at the periphery is higher, so that may give you -- accentuate your positive feedback.

CHAIRMAN KRESS: Yes. That could be real excursion, couldn't it?

MR. CARLSON: So, well, that would be interesting to see what kind of excursion it gives you.

CHAIRMAN KRESS: That won't take place for two or three days, right?

MR. CARLSON: That's right. That's right.

And, obviously, that's one where you would need spacial kinetics to do it properly.

The issues of worth of reactivity control and shutdown absorbers, there have been experiments done in recent years to help validate those calculations, and it remains to be seen what kind of tests will be done in the first modules of the designs.

We already discussed moisture ingress reactivity. Reactivity transients, I'll discuss that a little bit more later, but that's an important issue in terms of what kind of testing needs to be done on the fuel.

There's little or no in-core instrumentation. In a pebble bed, there are no structures to accommodate in-core instrumentation, and even in a prismatic design the temperatures are too high to allow much instrumentation. So that gives you issues of what can you do with ex-core instrumentation, and that's clearly a nuclear analysis issue that will require careful consideration. Clearly, the lack of in-core instrumentation may leave you with some uncertainties in terms of how far you can go in validating your nuclear analysis methods.

MEMBER SIEBER: I would imagine doing a

calimetric on a pebble bed would have more uncertainty than you would out of a ranking cycle.

MR. CARLSON: So there would be uncertainties overall in the thermal power is what you're saying.

MEMBER SIEBER: Right.

MR. CARLSON: I haven't really considered that. That's a good point.

MEMBER SIEBER: Well, but that's how you calibrate your ex-core instruments. So you're sort of out in there a little bit of no-man's land, a little bit.

CHAIRMAN KRESS: MC sub p, delta P.

MEMBER SIEBER: Pardon?

CHAIRMAN KRESS: MC sub p, delta P.

MEMBER SIEBER: Yes, but because you don't have heat of vaporization in there, you have to really know what the flow is --

CHAIRMAN KRESS: The flow is pretty close --

MEMBER SIEBER: -- and the temperatures.

CHAIRMAN KRESS: -- to delta p.

MEMBER ROSEN: Why is that a challenge, Jack? I mean you can measure the flow, can't you? You can measure the delta p pretty accurately.

MEMBER SIEBER: But the spread is not real big. The difference between a primary calimetric and a secondary calimetric. It's the heat of vaporization that really gives you the accuracy there. And it's 1200 Btus.

MEMBER ROSEN: What is the core delta p typically on these machines?

CHAIRMAN KRESS: Nine hundred minus 600, I think.

MR. CARLSON: About 300, 350.

MEMBER SIEBER: It's 200 to 300 degrees.

CHAIRMAN KRESS: Something like that.

MEMBER ROSEN: Sounds like enough to measure.

MEMBER SIEBER: Well, I think you can measure it. The flow is the tougher one, because it's a pretty light density material.

MR. CARLSON: And during Joe's talk, we mentioned the graphite and helium heat sources, although the graphite is operated at temperatures so that you don't get a significant accumulation of wigner energy; that is, continually. There are some higher energy graphite distortions that accumulate, and those only anneal during accident heat-up events. And that's an exothermic annealing so that becomes a heat source that you add to your decay heat source term. And, actually, the convention wisdom is that the dominant effect is that you recover some thermal conductivity in the graphite.

CHAIRMAN KRESS: You should note that you're giving this talk and Dana is here.

(Laughter.)

MEMBER ROSEN: So it's a good thing, right?

MR. CARLSON: I'm not saying I have concluded that, but others have concluded that the dominant effect is the recovery of thermal conductivity.

Some unique issues to the GT-MHR, in addition to fissile particles that are 19.9 percent U2-35, you have fertile particles that are natural uranium, so that's a unique challenge for modeling and validation right there. Also, burnable poisons and the zoning of fuel and poison loading is to give you the power shaping to limit peak powers.

For the PBMR, you have a very different core. You have a random loading of pebbles and continuous online loading where you measure the burnup of each pebble as it comes out and either put it back in the reactor of discharge it, depending on what the measured burnup is. The target maximum burnup for the PBMR is 80 gigawatt days per ton, so at what measured burnup do you discharge? And that becomes a question of how much additional burnup can you get on that last pass through the core, and that's a question of what's the residence time spectrum of pebbles on the final pass through the core?

I think one issue that the PBMR --

CHAIRMAN KRESS: How will you ever get that information, because it will depend on the level of burnup or the level of irradiation that the pebbles experience. And the way you're going to test that is with fresh pebbles somewhere outside to see what --

MR. CARLSON: What the residence time spectrum is?

CHAIRMAN KRESS: -- the residence time is.

MR. CARLSON: Well, actually, in AVR, they've got a pretty good measurement of residence time spectrum, and they did somewhat in THTR just by --

MEMBER SIEBER: That's the distribution, though, right?

MR. CARLSON: That's the distribution.

CHAIRMAN KRESS: You'll have to treat as a distribution.

MR. CARLSON: It will be a statistical argument, yes.

And there's an issue of using the four percent enriched fuel in the initial core, and do you really want to drive that to 80 gigawatt days per ton, and I don't think that's an issue that the PBMR design team has grappled with. My guess would be that you would want to discharge those at a lower burnup, but you can't distinguish between what the initial enrichment of a pebble is by measuring its burnup.

CHAIRMAN KRESS: That's right.

MR. CARLSON: So I see a bit of quandary here.

Some of you may have heard about the hot spots issue. I worked in Germany, and the AVR reactor was outside my window when I worked there for five years. One of the experiments they did there was a melt-wire experiment where they loaded 200 graphite pebbles, graphite only, no fuel in them, with melt wires, 20 different melt wires. The maximum melting temperature of the melt wires was 1280 C. And what they didn't expect was to get all those wires melting in any of the pebbles, but what they did in fact see was that ten to 20 percent of the pebbles had all the wires molten, indicating that the maximum coolant temperature, not the fuel temperature, the maximum coolant temperature seen by the pebbles was over 1280 C. So that's the hot spots issue, and it's not resolved. Perhaps the bottom line is that any new pebble bed reactor that's built will have to do melt wire experiments or something equivalent to that, both for the initial loading and perhaps the transitional and equilibrium cores as well.

MEMBER SIEBER: It's not clear how you would solve the problem, though, once you recognize that it was there.

CHAIRMAN KRESS: But you have to deal with it like we do the hot fuel channel in the LWR, treat it like the --

MEMBER SIEBER: Operate below the --

CHAIRMAN KRESS: Yes. You have to have some criteria for the hot spot.

MR. CARLSON: Just as a side note, when Exelon and the PBMR design team presented to us in June of last year, they were saying the maximum fuel operating temperature in the PBMR would be, what was it, 1100 --

CHAIRMAN KRESS: Twelve hundred.

MR. CARLSON: -- less than 1200. I think it was going to be 1060 --

CHAIRMAN KRESS: Yes.

MR. CARLSON: -- for maximum outlet temperature of 900 C. And I just said, "Did you consider the results of the AVR melt wire experiments?" And their answer was, "Not really." And I guess at our last meeting with them where we discussed this, they were saying the maximum fuel operating temperature is now 1300 C, something like that. And still nobody knows, and they won't know until they do a melt wire experiment or something like that in the first module.

MEMBER SIEBER: Even those aren't really the maximum temperature, right? It's a non-fuel ball.

MR. CARLSON: Yes.

MEMBER SIEBER: And so some fuel ball is going to have the maximum temperature.

MR. CARLSON: The best that can do is tell you the maximum local coolant temperature in the core.

MEMBER SIEBER: Right.

MEMBER ROSEN: It doesn't tell you that either. It tells you the maximum measured molten fuel.

MR. CARLSON: Yes.

MEMBER ROSEN: There may be a pebble that wasn't measured that was hotter.

MR. CARLSON: Maybe 200 melt wire pebbles isn't enough to give you a good sampling.

And then there are a number of issues of analytical treatments of the quasi-random local mixing of pebbles with different burnups, different fission powers and different decay heat powers.

MEMBER SIEBER: Do we know the degree of randomness of the distribution of these spheres?

MR. CARLSON: I would say no. I don't think there's been ever a direct way of measuring what is the clustering of first pass pebbles.

MEMBER SIEBER: Straight through the middle or --

MR. CARLSON: Well, there have been experiments done, and there have been measurements done on operating reactors that give you the residence time spectrum, and it gives a velocity profile that the pebbles move faster through the center of the core than they do at the core periphery and those kinds of things.

MEMBER SIEBER: Well, I would imagine you would build up a lot of fairly high burnup fuel on the outside and all the stuff you're putting in with it down through the middle.

CHAIRMAN KRESS: But how do you load this?

You load the fuel in the annular region off the top.

MR. CARLSON: Yes.

CHAIRMAN KRESS: Is it put in kind of distributed across the whole thing?

MR. CARLSON: There are nine different loading tubes around the periphery.

CHAIRMAN KRESS: And you drop them right in the middle of the annulus?

MR. CARLSON: In the middle of the annulus. Well, I think they still have a porous central reflector, although that may go away. But if they have a pebble central reflector, then they have a single central loading tube for that, for those graphite-only pebbles.

CHAIRMAN KRESS: Those are graphite-only.

MEMBER ROSEN: And you've purchased a set of body armor for your discussion with the ACRS, the full ACRS later this week when Dana Powers is here, on this subject?

CHAIRMAN KRESS: I would recommend you sit over where Richard is.

MR. CARLSON: For the pebble bed mechanics issue, the net mixing and flow of pebbles?

CHAIRMAN KRESS: Yes, I would recommend you sit over where Richard is, because Dana will be sitting on that corner.

MEMBER ROSEN: He's been coiling up for about two years on this subject.

MR. CARLSON: Well, that was one of the interesting things we discussed when we visited Germany last summer was the lessons learned from the THTR. They had predicted a given pebble flow velocity profile, and what they got was quite different, because the tests that they had done were scaled room temperature tests in air.

CHAIRMAN KRESS: Yes. I think what you'll hear from Dana, though, is he'll say, "Right on. You've got the right issues, you're thinking right." So I don't think Dana will be given him any problems. He'll just be saying, "Yes, yes, you've got the right idea."

MEMBER SIEBER: Well, it's a question, and I guess that that's the idea you ought to have, right?

CHAIRMAN KRESS: Yes.

MEMBER SIEBER: Instead of making an assumption.

MR. CARLSON: And in addition to the nuclear analysis issues directly for reactor systems analysis, there are some nuclear analysis studies that are needed to support the TRISO Fuel Testing Program.

The first one, as we alluded to briefly before, was reactivity transients. For defining the accident testing requirements, we need to define the worst case power transients that could arise from a credible reactivity accident, like a prompt pulse in a given HTGR design. We conclude that promp pulses are credible, we should try to consider the appropriate pulse width in addition to the energy distribution. There has been some pulse testing of fuel done in Japan and Russia, but to my knowledge, they used pulse widths on the order of ten to 30 milliseconds. Whereas in a graphite-moderated reactor, the real pulse widths are more on the order of 500 milliseconds.

CHAIRMAN KRESS: How do you get a prompt pulse in a graphite reactor? Do you have reject a rod?

MR. CARLSON: You'd have to reject a fairly high-worth rod or a bank of rods.

CHAIRMAN KRESS: That's about the only way I can think.

MR. CARLSON: Now, people have discussed pebble bed -- seismic compaction of a pebble bed --

CHAIRMAN KRESS: Oh, yes.

MR. CARLSON: -- as a way. The German

analysis concluded that you could only get about a little over one percent compaction. The theoretical compaction you could get would be over ten percent,in which case that would be well over prompt critical.

And also we have out-of-pile accident testing. The heat-up testing that Stu referred to and the pulse testing that has been done to a limited extent in Japan and Russia were done after irradiation with some time interval between irradiation and testing of days or months even.

CHAIRMAN KRESS: Yes. That's always the case.

MR. CARLSON: And that's the same for light water reactor fuel, and there has been an issue with that. So a similar issue applies. We need to do some nuclear analysis to evaluate how the radionuclide decay and other physical changes that occur before out-of-pile accident testing affect the radionuclide inventories that affect fuel performance in those accident tests. And, of course, the physical changes would be things like chemical reactions and phase changes.

Then, finally, for the irradiation in test reactors versus HTGRs, since most of the fuel irradiation testing has been done in test reactors rather than HTGRs, we need to consider how the radionuclide inventory, as it affects fuel performance, are affected by the non-prototypicality of those irradiation in terms of the accelerated burnup rates and the non-prototypic fuel temperature histories, the neutron fluences and the neutron energy spectra.

The rate of plutonium production and the ratio of plutonium fission to uranium fission is known to be pretty sensitive to neutron energy spectrum. So those are the kinds of things we would look at. The yield of significant fission products that is significant to fuel performance from plutonium fission versus uranium fission is significantly different.

CHAIRMAN KRESS: Yes. Now, when you say this is something you have to look at, you know, you've got the codes, you've got the cross-sections, and what I envision these tests in, say, the test reactors were just a way to validate the code predictions, how well did the code predict that. And then you say, okay, my code has the right cross-sections and stuff, so I can predict an actual HTGR because I know the cross-sections of plutonium, and I know the energy spectrum I'm going to get is going to be different, but I can account for it. What do you mean when you say you're going to look at? You're going to do more --

MR. CARLSON: We do some calculations. Let's take an irradiation in HFR or the ATR or the HFR.

CHAIRMAN KRESS: You do it in a variety of reactors that you can.

MR. CARLSON: And calculate the spectrum that the fuel sees in those tests. Calculate the spectrum that you see on actual HTGR --

CHAIRMAN KRESS: Power those.

MR. CARLSON: -- irradiation. Take account to the accelerated burnup if you have that. And compare the nuclide inventories you calculate with one versus the other. If there are significant differences, then we should factor that into interpreting the applicability of the test results.

CHAIRMAN KRESS: Or the applicability of the code calculations. I view this just like thermal-hydraulic. You know, you validate them in non-prototypic conditions, but you figure the range of --

MR. CARLSON: Well, I don't think any of these tests validate the nuclear codes.

CHAIRMAN KRESS: You don't view them in that light?

MR. CARLSON: No. Their purpose is to test the fuel, and I think they have little or no value for validating the nuclear methods.

CHAIRMAN KRESS: Okay. Could they be used for that?

MR. CARLSON: You would have to retool -- they'd have to design the experiment to really get what you want for nuclear analysis validation. And there are facilities that are designed to really do that sort of thing.

Some of the research activities that we're starting or planning on soon starting for the GT-MHR and PBMR, the advanced HTGRs, number one, we're -- first, we've started to prepare modern nuclear data libraries based on the latest data evaluation files in ENDF/B-VI.

CHAIRMAN KRESS: Who is the custodian of that data?

MR. CARLSON: Brookhaven.

CHAIRMAN KRESS: Brookhaven.

MR. CARLSON: Brookhaven is ENDF/B-VI custodian.

CHAIRMAN KRESS: Okay.

MR. CARLSON: Back in '96, when I was in NMSS, I initiated a user need for research to update the Apex code system, which is a code system at Oak Ridge that is used to process the evaluated nuclear data from ENDF/B-VI or the foreign counterparts, JEFF-3 or JENDL-3.2, into actual cross-section libraries. And that's exactly what we've started now that -- in response to that user need, now that the Apex code has been upgraded to do that job, and there's also the NJOY code at Los Alamos that can do part of that job. We're going to use those tools to generate state-of-art cross-section libraries to ultimately replace the libraries that are in use today in the NRC, which are mostly from the 1980s and based on ENDF/B-IV and ENDF/B-V.

So we're talking about multi-group libraries with perhaps 400 to 500 energy groups that would generically applicable to all reactor types, not just HTGRs, including current generation light water reactors and would be used for all in-reactor and out-of-reactor nuclear analysis applications.

MEMBER SIEBER: Just for my own education, what do we know now about ENDF/B-VI data that we didn't know in version III or IV?

MR. CARLSON: There's a whole list --

MEMBER SIEBER: Is it new measurements?

MR. CARLSON: There are some new

measurements. There are improvements in the tools used by evaluators when they take those measurements to connect the points, so to speak. Significant improvements there. There have been plain glitches that have been caught. I had a hand, some 11 years ago, in catching a problem in the S-alpha/beta bound thermal scattering data in ENDF/B-VI and actually had gone back to ENDF/B-I. And it was particularly significant for graphite.

MR. LEE: And also in the Apex code, the suite of codes that we developed, the residence treatments are better now, either in the resolved or unresolved residences. So those tools have been developed now, so we need to process the data to get these cross-sections for application.

MR. CARLSON: The ENDF/B-VI formats greatly increase the resolved energy range, the resolved residence range for the data.

MEMBER SIEBER: Do you see improvement in the use of that in 3-E diffusion calculations as far as accuracy of predictions or --

MR. LEE: I think in our recent staff application in the, for example, the peach bottom turbine trips --

MEMBER SIEBER: Okay.

MR. LEE: -- one, the reactivity assertions you can see some difference between applying the two different type cross-sections.

MEMBER SIEBER: So it's a worthwhile endeavor to do this.

MR. LEE: Yes, definitely.

MEMBER SIEBER: Okay. Thank you.

MR. LEE: Across the board.

MR. CARLSON: And it shows up in the depletion analysis and in shielding calculations everywhere.

Also, we're starting scoping studies for core neutronics and decay heat analysis. The general approach is to use high-order methods, like continuous energy Monte Carlo, NCNP, and do very exact models with exact geometries and gradually introduce the approximations and more approximate methods that are used in practical reactor analysis codes to understand what the effects of these approximations are and what would be acceptable modeling practices and their range of applicability.

We've initiated some PARCS code modifications to incorporate an R-theta-Z geometry that would be needed for analyzing a pebble bed reactor.

MEMBER SIEBER: Okay.

MR. CARLSON: And we envision some PIRT exercises that would be focused on the reactor systems analysis area, including nuclear analysis to identify and more systematically prioritize the particular needs to data and modeling capabilities.

We're also planning some cooperation with MIT on a core depletion analysis tool that would build upon the peb bed code that's been developed in conjunction with INEL. And we're pursuing opportunities for HTGR-related domestic and international cooperation to get access to physics benchmark from various sources. We'd be going first through the IAEA. There's a cooperative research program, Number 5, that's been ongoing since 1998 and scheduled to go through 2004. That has been looking at the initial criticality and physics data from the HTGR in Japan and the VHTRC critical -- the heated critical experiment facility there; also, the HTR-10 initial criticality and subsequent benchmarks from China; the Astra Facility at the Kurchatov Institute in Russia that has been -- those are pebble bed experiments with in-reflector absorbers that have been sponsored by PBMR.

And then the HTR PROTIS experiments from

PSI in Switzerland that were done in the early '90s. That was an international program. And, finally, some data from France, Germany and the U.S. and U.K. I was involved when I was at Los Alamos in the CNPS critical experiments, and those would play a role.

In addition, as part of the international cooperation, we're considering providing U.S. NRC assistance, both in the technical aspects of the testing programs but also in the QA areas to make sure that the quality assurance is adequate, that we can actually make full use of the results from the testing programs.

So now that concludes the nuclear analysis. I can turn it over to Richard.

MR. LEE: Starting with the AP1000, as you know, this application is in-house, and NRR is planning to issue a draft SER sometime in June of next year, following with a final SER by the end of fiscal year FY '04. Related to the AP1000 back in February 14, the research and NRR staff has briefed the Subcommittee in detail about the AP600 scaling and how it is applied to AP1000. And I think you know a lot more about AP1000 thermal-hydraulic analysis requirements for this application in details.

As you know that the -- we said that most of the work that we have done in support of AP600 means the Apex facility at Oregon State University, all those tests are applicable, accept that we believe that the range and some of the conditions need to be extended for applicability to AP1000 and mostly related to the steam production, high-costing production that resulted in high entrainment for horizontal stratified flow and the upper plenum pool entrainment. Both experiments are ongoing at this time.

MEMBER SIEBER: Who's doing those, Westinghouse?

MR. LEE: Westinghouse is doing the integral effects. They modified a facility --

MR. ELTAWILA: Correction.

MR. LEE: No, not that one.

MR. ELTAWILA: This is DOE testing, not Westinghouse.

MR. LEE: Oh, DOE.

MR. ELTAWILA: DOE.

MR. LEE: Yes. I should say DOE.

MEMBER SIEBER: But the entrainment issue --

MEMBER ROSEN: That's done at Oregon State.

MR. LEE: Yes. It's been done at Oregon State at the integral facilities.

MEMBER ROSEN: With DOE funding.

MR. LEE: DOE funding, right. What NRC is doing with that, before they change configuration, there are some certain other conditions that we'd like to test. Those tests are sandwiched between the DOE testing. And I believe we are also doing some separate effect testing, looking at the entrainment phenomena details.

MEMBER ROSEN: All of this will support the 2004 SER?

MR. LEE: Yes. I think even before that. I think by beginning of next year I believe that we need to get our codes in shape.

MEMBER WALLIS: This entrainment from horizontal flow, what is that?

MR. LEE: I think it has to do with the Ts.

MEMBER WALLIS: That's the ADS for T.

MR. LEE: Yes, that's correct.

MEMBER WALLIS: So it's entrainment at a T, really.

MR. LEE: As a T; yes, that's correct.

MEMBER SIEBER: Well, it sweeps across the top of the core and carries water that's supposed to be cool in the core out of the break.

MR. LEE: That's right.

MEMBER WALLIS: Carries it out the ADS fall line.

MR. LEE: It's the ADS fall line that we're talking about, right, and the concern about --

MEMBER SIEBER: And that's different because the ADS system is different between the 600 and 1000.

MR. LEE: That's correct. Right. Especially ADS. Those are ongoing. Then another thing to talk about is the low pressure critical flow. We are doing some testing at the Purdue University, and that is basically to look at much lower found in 150 psi regions for critical flow. They are mostly at the high pressure. This ECCS bypass direct vessel injection, those are being looked at, the data from Korean's program.

For the IRIS reactor, as you know, that the steam generator pressurizer cooling pumps, everything is located inside.

CHAIRMAN KRESS: What is meant by modular in this sense? Is it the components are modular or you have modules of reactors?

MR. LEE: I think it's a small unit, so I guess they can build --

CHAIRMAN KRESS: Use three or four of them to get 1,000 megawatts? Because some people speak of modular as the parts are modular that go into --

MR. FLACK: It could be also modular, but in this case they're talking about the reactor themselves as being modular of anything more than one site.

CHAIRMAN KRESS: More than one.

MR. FLACK: You have several of them to a site.

MR. LEE: Right. You can see that the power is about this much. And the size of the whole vessel is about 60 feet tall, so it's about almost two times the height of a current reactor, the pressure vessel.

The issues that we look into of course has to do with -- the steam tubes that they use are different than current design, because this promotes very good T transfer because of the heat transfer. Then the reactor also relies on a lot of natural circulation. About 40 percent of the core flow are driven by natural circulation during an operation. And then another thing is that the way that the -- if anything happens, the RCS gets depressurized to a very low pressure and close coupling between the containment and the RCS, just like passive reactors we have now. For the SBWR or the AP1000, there's a close coupling between the containment RCS.

CHAIRMAN KRESS: You're going to have to hook -- does MELCOR core already have that coupling in it?

MR. LEE: We're not doing anything right now on it, but, yes, we do have the containment and the --

CHAIRMAN KRESS: I guess the new track M would have to be connected to something like contained to evaluate the thermal-hydraulics for the strong coupling between the containment and the primary system?

MR. LEE: Yes.

MR. ELTAWILA: Yes. Right now, the TRAC-M code has a very simple containment model, so you can use it. But the long-term plan is to couple the contain code to the TRAC-M code.

CHAIRMAN KRESS: That's all you're really looking at is the back pressure effects on the blowdown, which you could use a simple model for that thing.

MEMBER WALLIS: Blowdown is from a steam line break; is that what it is?

CHAIRMAN KRESS: Yes. You've got the steam -- that's the only place --

MR. LEE: That's the only one coming out from this reactor. That's the only thing that is coming in is the stream generator feed and the one going out.

CHAIRMAN KRESS: A small volume, strong containment, so that it builds up in pressure pretty fast.

MR. LEE: Right. It's a very small containment.

CHAIRMAN KRESS: So it affects the blowdown rate. That's probably the only thing it affects, I'm not sure.

MR. LEE: Right. And then as you --

MEMBER WALLIS: The primary water can't get out?

CHAIRMAN KRESS: Well, that depends on whether you have a steam generator tube rupture, I think.

MEMBER SIEBER: It gets out.

CHAIRMAN KRESS: I think that's the only way.

MR. ELTAWILA: According to Westinghouse, you can run with a LOCA forever.

CHAIRMAN KRESS: That's with water. But I think you have to rupture the steam generator tubes to get water out, unless you can get a break in the vessel itself, which is --

MEMBER SIEBER: That's right.

MEMBER WALLIS: You presumably have smaller breaks. You presumably have make-up water for the vessel or something. You must have some lines.

CHAIRMAN KRESS: Well, you may have control rods going in. I don't know what the penetrations are, but you may have some control rods.

MR. LEE: The control rod guide tubes are coming in from the top, but my understanding is that those can be even relocated into the vessel. That's what we mentioned.

MEMBER WALLIS: So those can break. Those can break, even after you solve the problems we have with the control rod.

MR. LEE: That's one.

CHAIRMAN KRESS: You may have to rupture the head to get a leak.

MR. FLACK: Actually, we have somebody from Westinghouse here that can speak. You can use the microphone.

MR. ORIANI: I'm Luka Oriani from Westinghouse Science and Technology Department, and I'm working on the IRIS design. We actually are considering some intermediate and medium-size LOCA because we will have some piping. For now, the assumption is that the largest piping will be a four-inch pipe, more or less.

There are also some differences in the design with respect to the considerations that have been presented here. Like, for example, the degree of natural circulation is much lower. That 40 percent was referred to is more a size of the IRIS reactor that was initially foreseen, and the parallel channel flow instabilities should be less of a concern, because the core thermal-hydraulic design is pretty much straightforward. And those are from the neutron analysis point of view.

The enrichment is a standard enrichment. It's below five percent, and the fuel cycle we are going to decide in the next few weeks between two remaining options. One is for a four-year straight burn cycle, and another one is for fuel shuffling on a three-year cycle.

CHAIRMAN KRESS: It's almost impossible to get of the instability region because of this natural circulation.

MR. ORIANI: Actually, natural circulation in operation will not be terribly different from other light water reactor. It will be a higher degree of natural circulation, but it's not 40 percent as it was initially foreseen for different sizes. But two-phase natural circulation becomes important, especially in LOCA events and in those kind of accidents.

MEMBER SIEBER: This is the reactor that had the primary coolant on the shelf side of the steam generator?

MR. ORIANI: That is correct, yes. That's also the reason why the steam line break actually doesn't lead to a release of mass flow containment, because there's no mass inside the steam generators.

MEMBER SIEBER: Okay. Thank you.

MR. ORIANI: You're welcome.

MR. LEE: So as you know, the design itself is, as we mentioned, what we've written here, the information that's provided to us. Based on that, this was written.

As any other advanced reactor, we think we need to have integral as well as separate effects to validate our the model codes. And the integral ones, of course we talk about the containment-RCS coupling. The separate effects we like to see how the steam generator performs under normal as well abnormal conditions. There are a lot of design -- chemical and process industry has a lot of data on the core steam generator, but we expect that the size of this and the conditions that are going to be operating will be different, so we need to examine the performance of the steam generator under the condition that we are looking at.

MEMBER FORD: Now, as I understand it, there's other work going on on advanced light water reactors. There's a thermal-hydraulic link --

MR. LEE: Yes.

MEMBER FORD: -- on the SBWR and SWR-1000.

MR. LEE: That's correct. the ESBWR, yes, we are -- we're going to be supporting, as Farouk has mentioned earlier in the morning, the ESBWR design certification. So we are going back to the time in the early '90s when we terminated the SBWR review. We're going to start from that point and pick up and look at what the issues that we need to look into.

MEMBER FORD: And this is related to melt retention issues?

MR. LEE: No. This is -- to begin with,

we're going to be in the thermal-hydraulics related issues that we're looking into, but now we have to look into the scaling that we have done at that time and scale it back up to the higher power that the ESBWR expected to be.

MEMBER FORD: This thermal-hydraulic stuff is related to work to be done at the PUMA facility?

MR. LEE: Yes, that's correct. So we have done some work at Purdue already, so we'll use that as the starting point.

MEMBER FORD: So the fact that you don't have this in this presentation, where you're just talking about the MHR, the gas cooler reactors and the AP1000 and IRIS, does that mean it's being funded in a separate -- it's being considered in a separate program or is it within this program?

MR. ELTAWILA: As you recall, this plan was developed in February when PPMR and the AP1000 are the two programs that plants were reviewing. GE came in June of this year. So as a result of their decommission, asked several questions about what are the resources that we needed. So we itemized some resources to the Commission, and it's between now and August a decision is going to be made at the Commission whether to fund it from the existing program or request supplemental funds from Congress to address this issue.

But regardless, as I mentioned early today, since some of the heat related to gas cool reactor has been delayed, we are reprogramming some of 202 money to start doing some ESBWR work. So it's going to be funded, there is no doubt about it, but the question is will it be funded as part of the budget that approved by Congress? Because the '03 budget has been approved. So anything above that we have to go to Congress for supplementary funding.

MEMBER FORD: The reason why I ask the question is just as we go down this whole list for the plans you have, you're going to have prioritization issues and how you're going to allocate your monies, and I heard you talk about --

MR. ELTAWILA: That's correct, yes.

MEMBER FORD: -- this particular thing. Okay.

MR. LEE: And beyond this, we're also looking into CANDU Reactor as well, the ACR --

MR. ELTAWILA: Seven hundred, yes.

MR. LEE: -- 700, yes.

CHAIRMAN KRESS: Richard, could you go back to the previous slide? I had one more question on that. On your last bullet, do you actually envision a Rosa-type of an Apex-type facility for IRIS?

MR. LEE: For integral facility, that's what we're thinking about, yes.

CHAIRMAN KRESS: Yes. That one bothers me a little, because --

MR. LEE: I don't know whether --

CHAIRMAN KRESS: -- IRIS doesn't really have any ECCS like the standard. It's got all the water in there already, and the questions you had with these other facilities is can you actually get the stuff in there to the core to keep it cool? And really all you're dealing with with IRIS is what are the blowdown rates, and you don't have to have a full integral facility to determine blowdown rates. So, you know, I'm questioning whether there's a need for Westinghouse to build a full or even scaled facility with electric rods in there for an IRIS-type facility, because the design is such it looks like you don't really need that kind of detail. Am I wrong there?

MR. LEE: No, but there is a natural circulation time that the water in the containment will be circulating through the vessel and removing heat from the vessel.

CHAIRMAN KRESS: So you think --

MR. LEE: So how this very small delta p between the containment and vessel is going to actually cause the circulation and with changing in temperature and all this stuff you really need to understand how it's going to work. So although you might not have -- the blowdown itself is not the issue as much as the processes between the vessel and the containment after the LOCA itself.

MR. ELTAWILA: And, again, as Richard indicated, we really don't have enough information about the design to make a judgment at this time. But we're saying if this design is going to be radically different from what we have learned in the past, we might require a test facility. So a decision has not been made that we are going to build a facility.

CHAIRMAN KRESS: Yes. I would think about that one long and hard, because --

MR. ELTAWILA: No, I appreciate this.

MR. LEE: And I expect that we're going to use a process to look in all the phenomena before we do anything on this, even though it's not mentioned here.

And then back to the gas cool reactor, and we know that the fluid flow and heat transfer here are different because they are different medium. The code as I mentioned to you is that -- and using the TRAC-M code and then if needed we will use the FLUENT to look at more details, if there's any specific thermal-hydraulic issues that we have to look at. As you know, TRAC-M doesn't have the -- I mean, we need to put the helium, we need to put the carbon as graphite as a solid structure. For the PBMR, we need to put the spherical fuel in there. And then for the turbo-machinery, I think we do have models. We need to extend it to the different types of energy conversion device. And then on the passive heat decay removal system, whatever is going to be used, we need to modify those.

Into the severe accident arena, we are also supporting NRR in this -- supporting on this phase two design certification, and you remember that we don't expect a severe accident source term to be different between the AP1000 and the 600. I mean it's the same design, but after AP600 design certification was completed, NRC has done some more experiments at the OECD Rosecroft and Masco. We learned something from there on the in-vessel melt behavior. Those knowledge we need to be transferred for the application to the AP1000.

CHAIRMAN KRESS: As best as I recall, it was barely adequate for the AP600.

MR. LEE: That's correct, for the in-vessel retention plan.

CHAIRMAN KRESS: When you go up to 1000, you've got a lot more decay heat to deal with.

MR. LEE: Right. You have two issues. It has to do with in-vessel melt behavior, how does the heat flux distribute between the bottom head and the site on the spherical hemisphere. Then another issue has to do with the external cooling with water, and the experiment that we have done for AP600 at that time was at Penn State and USC-Santa Barbara. Those experiments showed that the critical heat flux -- the margin between the critical heat flux there's some margin there.

Now, with the higher power density now, that margin has been eroded. But we also understood that at USC-Santa Barbara, they're doing some more work by redesigning the insulation outside of the hemisphere. Essentially, what he's trying to do is to increase the critical heat flux by forcing the flow going up so try to regain some of those margins, but we haven't examined those data yet, so we have to look at those closely.

CHAIRMAN KRESS: But let me ask you a hypothetical question.

MR. LEE: Yes.

CHAIRMAN KRESS: Suppose AP1000 comes up with that this was marginal and that they don't want to take credit for it in their safety case because it's too marginal, but they say, "But we're going to do it anyway. We're going to flood the vessel anyway. We're not taking any credit for it in our safety case." Does this reopen, in your mind, questions of steam explosions?

Because now you have water there ready and you have a melt. It might go through the bottom head, and it's probably separated with the metal phase on the top where it penetrates. That's where the vessel fails first. So you've got to relatively medium pressure in there blowing out liquid metallic components into water that's already there. Does this, in your mind, raise the possibility of having to relook at steam explosions?

MR. LEE: Research is looking into -- if the in-vessel retention doesn't work and if the pressure vessel fails, we are looking into the so-called ex-vessel phenomenon. That includes the FCI, DCH, hydrogen combustions and all those.

CHAIRMAN KRESS: Do you think --

MR. LEE: But remember Westinghouse said if the in-vessel retention fails, they assume containment fails. The probability is one. That is the argument now being forwarded, yes.

CHAIRMAN KRESS: Yes, okay.

MR. LEE: In the PRA analysis.

CHAIRMAN KRESS: Yes. I remember --

MR. LEE: But, nevertheless, NRR requested us to look into the external FCI, all those issues, yes. So that's why I said at the last bullet.

For this reactor, the design is not fixed yet, so the -- I think our discussion is that the fuel doesn't look that much different to us or we said the progressions and all those core issues be that much difference between IRIS and light water reactor. That is my opinion.

CHAIRMAN KRESS: I guess I would --

MR. LEE: That's my opinion.

CHAIRMAN KRESS: -- have to question that. We've got much higher burnup, we've got all these burnable poisons in there. We've got a slower heat uprate because of the decay. You know, it took longer to get to the meltdown. I think I would expect the meltdown and fission product release processes to be considerably different from what we're used to.

MR. LEE: The higher burnup is up to around 80, so we are now looking beyond around 65, 70? So --

CHAIRMAN KRESS: Yes, but we don't even deal with 65 hardly. The database for the fission product release is obtained from around 45 gigawatt days burnup. So, yes, I would expect the meltdown and fission product release to be a lot different for IRIS.

MR. LEE: And as you can see that right before we do anything we're going to start another process to find out what we have to do for this design once the design is fixed.

Now, I have to say that the fission transfer to the primary system we need to look at it in even more detail now because of the -- the steam generator is different. So we are going through a very troubling deposition inside the core, and we don't have those models for transfer for that type of steam generator. So we expect that the fission transfer to be different.

MR. ELTAWILA: But, Tom, you heard the presentation a few months ago from CAIRSN in France, which they are planning to run some REBUS 2K test to look at high burnup fuel. So if the fission product release and the core melt progression looks any different, I know it's a very small experiment and things like that, but once we see this information we'll determine whether really the core melt progression is going to behave differently for high burnup fuel, and at that time, we'll revisit the issue. But there are some work that's going to be done in on high burnup fuel. And we are going to be part of that program.

MR. LEE: And the French may even conduct a fission product release test for up to like 75 gigawatt days per ton.

CHAIRMAN KRESS: Yes, I understand they're going to do that. Are they going to include these burnable poisons?

MR. LEE: No, not that. Now, turning back to the HTGR, as you said, the sequence fission product release transport is expected to be different. Now, we have different few designs, either spherical or an prismatic design. And there are some other reactor internal structure that we have to take into account. For example, the graphite, for example, how would the deposition of aerosols interactions with graphites? I don't know the database on that, but we're looking into it.

We have initiated MELCOR development for the HTGR. It's for the base on the TRISO fuel, so you can use a spherical one or prismatic-type reactor. As Don mentioned, the code that has been used at Oak Ridge back in the '70s until the '90s, right, there's code here. And whatever we learned from then the modeling aspect has been used for thermal-hydraulics as well as for MELCOR, because the bases start from the same point. So we are taking into account what we learned from that.

CHAIRMAN KRESS: As best I remember, GRSAC doesn't have a fission product release model.

MR. LEE: Right.

CHAIRMAN KRESS: It just has thermal-hydraulics.

MR. LEE: So we're taking the thermal-hydraulics, but they may have some other oxidation models and so forth.

CHAIRMAN KRESS: Yes, but --

MR. LEE: And we're taking those, yes. But the fission product release model is still based on the MELCOR, the root diffusions. So, basically, at early morning you mentioned about what you envision for the MELCOR code. It's the same thinking that we are pursuing.

CHAIRMAN KRESS: But that bothers me too, because the --

MR. LEE: But you need to have a database.

CHAIRMAN KRESS: Yes. You have to have a database for that. And I envision the fission product release would be driven by how rapid these TRISO pellets fail. And that's a different concept than the fission product release models in MELCOR are -- it's based on thinking that it's a diffusion process, and I don't know if failure of these TRISO pellets has anything to do with diffusion. So even the concept of using the type of models, even though they are empirical in MELCOR, is even relevant for the HTGR.

MR. LEE: But at this time, that's what we're thinking about. But you know that this --

CHAIRMAN KRESS: You're going to need a lot of data.

MR. LEE: There's a fuel PIRT that's going on that we follow very closely, because the fission gas release and so forth start from the fuel because the barrier now moves from the cladding to the fuel itself. So we are following that one. And I think beyond that there will be some more discussion on how do we model the fission product release.

MR. CARLSON: I think there are some fission product release models in the old MORECA and the newer GRSAC code, and we'll have to look at how appropriate those are for the --

CHAIRMAN KRESS: I think they were for the actual fuel if they use the cladding. The gas cool reactor fuel at one time had cladding, and I think it was -- the release models were for that, but I'm not sure.

MR. CARLSON: We're working with GRSAC right now to exercise the models that are in there as they relate to TRISO fuel.

MR. LEE: As we mentioned, just like in other programs in the fuel, in neutronics, we are looking at all the other research that are done outside of this country at the HTGR research, in specific, Germany, in Japan and IAEA. IAEA has done many -- conducted many specialist meetings on gas cool reactors, and I think we are reviewing and see what is applicable from those studies.

I think earlier they mentioned about the European Commission on the HTGR research. We are planning to participate in those, and that is in like the fuel and in all the materials, and this is another area that we are looking into. Because they want to do some fission power release in the PIE on new experiments. So --

MEMBER SIEBER: Is that $16 million for the federal program or for --

MR. LEE: I think it's --

MEMBER SIEBER: -- our share?

MR. LEE: -- $16 million that they budgeted on --

MEMBER SIEBER: Is it total program funding and then we'll pay some share of that?

MR. LEE: I don't know. The U.S. participation may not have to put any money in.

MR. ELTAWILA: The way the European Commission they will not accept money, and they don't send money outside of the European communities. So in-kind contributions. So you try to do research in the same area and exchange data.

MEMBER SIEBER: Okay.

MR. LEE: So it could be our analysis in support of reviewing the program, what type of test could be appropriate to be conducted and so forth. Those are the type of exchange.

MEMBER SIEBER: Sounds good.

MR. LEE: So, in summary, in the reactor system analysis, we tried to capitalize on whatever ready access in internationally and then we are building basically on the LWR tools that we have developed to TRAC-M and MELCOR. PARCS is a kinetics code we develop at Purdue. Don mentioned earlier the lattice physics code that we developed at Oak Ridge, which is we are doing it for the MOX, but we can modify it for HTR applications. And that is part of the scale suite of codes at the NRC used for a lot of analysis, neutronics analysis. Then we also talked about expanding our capability to address new technology issues. That is in graphite helium, high burnup fuel, up to the 80's gigawatt days for IRIS reactors. That's all.

CHAIRMAN KRESS: Any questions of Richard? I guess we've asked them all. Okay. I guess you're going to wrap things up for us, John?

MR. FLACK: Yes. My plan was to summarize briefly the other technical areas and then summarize the entire meeting, you might say, and where we go from here.

CHAIRMAN KRESS: Will that summary be a good thing to present to the full Committee?

MR. FLACK: Well, we'll have to talk about that. But what did we hear so far? So we've seen the -- we've discussed in some detail the four technical areas, framework, skills, materials and reactor systems. And now I'll quickly go through the remaining technical areas, starting with the PRA.

As we look at these other areas, there's not as a radical change to the work that we're doing now, for example in TRISO fuel where we need to understand a new technology. A lot of the work in these remaining areas build on what already has been done, and it becomes more difficult to extend it unless we have a specific design in place. We talked about this earlier about being technology neutral, and at some point you need to have a plan. And so a lot of the remaining areas are, well, we could begin to understand or look at some of the issues that we can see, but really it's difficult to move further than that until you start to get a plant and apply it, apply your thinking process to that particular design.

But in the PRA, starting with the PRA area, of course we use PRA more and more since the PRA policy statement had been put forth in 1995. And, basically, there's three areas where we're using PRA. The first one and most importantly is to support regulatory decisions, risk-informed performance-based decisions in supporting policy issue resolutions and rulemaking to help resolve safety issues and to help identify uncertainties, the extent of those uncertainties and the sources of those uncertainties and Defense In-Depth and the safety modules.

Another use of PRA is to assess licensees' PRA. We need tools to do that. To some extent, we will certainly not be in a position to do our own PRA on a design as it comes in, but there may be certain facets of a licensee's PRA that we may want to look down into detail and may decide to develop the models further for our own use and seeing if we can their results.

And then, of course, we use PRA also in our research that we do and setting what are the priorities in the research that is ongoing and what needs to be done by identifying scenarios of risk significance and so on.

The technical issues, as we see them today, and a lot of this work, by the way, has been prepared by John Ridgely and Mary Drouin, and John is here to answer any questions that you may have on them. But I summarize these issues in the following five bullets. The initiating events were advanced designs, understanding what caused these initiating events that are different than light water reactor and the database that we can call upon to help us identify those initiating events. We see this as one of the technical issues we'll have -- challenges we'll have to come to grips with.

CHAIRMAN KRESS: You will need to pin down some sort of range of frequencies for those.

MR. FLACK: Yes. If we go back to the licensing approach that Exelon had used, for example, where they tried to allocate the events into different categories -- abnormal operating events, and then they had what was considered design basis events and emergency planning events. Yes, to the extent that we can, try to identify what the likelihoods of those events are and then, of course, the subsequent source terms it might be associated with.

CHAIRMAN KRESS: Yes. I never got a chance to ask them where the got those frequencies for those events.

MR. FLACK: Well, they probably got them from the MHTGR.

CHAIRMAN KRESS: Yes. I haven't gone back to see where they got them.

MR. FLACK: Yes, right. Where did they get them from?

CHAIRMAN KRESS: But there's not a large database like we have with a lot of reactors on what initiating event frequencies might be.

MR. FLACK: Yes.

CHAIRMAN KRESS: I just don't know where they got the numbers.

MR. FLACK: Yes. Some of it, of course, is you can probably draw from light water reactors.

CHAIRMAN KRESS: That's where I think they probably got them from.

MR. FLACK: Yes. But then there's others that it would be hard to draw from without large uncertainties.

MEMBER BONACA: Will you eliminate initiating events based on the probability alone? Say that you have a concern with a possible effect that seems to be of low probability. Are you going to eliminate that?

MR. FLACK: Well, I don't think -- you know, if we were in a risk-based arena, we might do that, but it's really -- of course, any probability has a distribution, and so one needs to understand the distribution making a decision. So there's always the -- the difficulty is that even -- and it's estimated and the probability is what's the technical basis for that probability?

And this gets into things that we've heard this morning -- or this afternoon about John Muscara's presentation on how we're going to resort to probabilities where there's limited data. So you are going to end up with large uncertainties. So the question is going to become -- it's going to come about, well, okay, is there a cliff somewhere where suddenly you go a little bit further and you have this large release of radioactivity.

A lot of the research that we do tries to really probe that question, and that's why we take things to failure. There may be enough margin, but then how much more do we go before we actually get ourselves in a problem? So I think the decision is going to be a combination of things when that time comes.

But, again, it is a challenge, and of course the challenge also is in modeling these different systems, confinement versus containment, and what credit one would give for something like this. And then passive systems are always difficult to quantify, recognizing the need to identify the failure modes of those systems and so on and the applicability of the data to advance designs, which you just discussed.

And then, finally, the human performance and a multi-modular design in I&C and how does that get quantified in the context of a PRA for an advanced design, and what is the role of a human in these advanced designs? So these we see as the challenges, basically in the PRA area right at this moment.

I don't know if John Ridgely wants to add anything to that? No. Okay.

MEMBER SIEBER: I have a question about the human performance. When you talked about the concept of modular designs, do you see one control room with a bunch of reactor control panels for each module or do you see those separated somehow or another? The reason why I ask the question I once worked in a coal plant with six units run out of one control room. If one unit would get in trouble, they would rush to that unit and the other ones would float off into never-never land until something tripped.

MR. FLACK: That's a good source of information. You know, part of the work -- actually, that leads me into my second viewgraph if --

MEMBER ROSEN: Let me just make a comment on the last bullet there. There is a risk in the Safety Cross-Cut Group in the GEN IV Program. The GEN-IV Program was divided up into gas-cooled reactors, liquid metal reactors, water reactors and advanced reactors or innovative reactors. Plus it had some cross-cutting groups. One of the cross-cut groups was the Risk and Safety Cross-Cut Group, and it identified that last bullet, the human performance modeling for advanced reactors as an issue also. And it's proposing that the DOE GEN-IV Program do some research work in that area. So you might want to make a note of that and look at what's going on there.

MR. FLACK: Okay. I think Steve Arndt actually has something to say about that.

MR. ARNDT: Yes, sir. We're quite aware that we actually participated in the workshop that they held about six weeks ago to develop those recommendations. And both our Human Factors and our RSC Group were very active in that actual participation in forming those research recommendations in coordination with putting this plan together.

MEMBER ROSEN: Good. Sounds like you're tied together.

MR. FLACK: Okay. And that sets me up with the next viewgraph, which is on human factors. And, again, this is simply -- this is the question we're asking ourselves: What is the role of the operator within the context of these advanced designs.

Under the normal operations, maintaining configuration and control, as well as accident response.

And, again, relying on I&C and automatic systems to perform a lot of the functions that operators perform today is going to be somewhat challenging as to if these systems fail to function under certain conditions where you are in a multi-modular design and one module is in one state and another is in another, and everyone's focusing on the one, and the rest of these are floating out there.

One of the efforts -- activities we're planning to do initially is to just do that, to go out into other fields and see what data is out there, whether it's cold units or others and see what kind of issues do come out of these multi-control room modular-type plants in other fields. So that's something we are planning on doing.

MEMBER ROSEN: The reliance on I&C I think refers to digital I&C?

MR. FLACK: Yes.

MEMBER ROSEN: Because all these plants will be totally digital by the time we get --

MR. FLACK: Yes. Right. That's right. In fact, we have another viewgraph that's going to -- you're leading me right into the next one. These are issues. But we think of advance designs being radically different for things like TRISO fuel particles. This is actually going on today. I mean we're seeing changes in current generation and some of the work that we would be doing looking at I&C on today's plants and it could change our control rooms actually carrying us right off into what we can imagine they'll be doing for advanced reactors as well. So we sort of have a foot in both ends there.

MEMBER ROSEN: Yes. I agree with you but only in part. I think there are a lot of limitations on what the kind of changes -- the digitization, let's call it, of the current fleet is very limited, by comparison, to what I understand we're talking about here, which are --

MR. FLACK: Where we're headed.

MEMBER ROSEN: -- six plants, one control room and one screen with the operator touch-sensitive screen where the operators hits which plant do you want to know about first. Now, that's the ultimate digitization.

MR. FLACK: Yes.

MEMBER ROSEN: Then you can drill down, that, that, that, that, that, that.

MR. FLACK: Yes.

MEMBER ROSEN: That's a completely different thing than what we're used to.

MR. FLACK: The question is, of course, how do you prepare for this before it comes in the door?

MEMBER ROSEN: That's why we've left that to you.

(Laughter.)

MR. FLACK: Appreciate that.

MEMBER SIEBER: In addition to the one screen, you need six lights to tell you which unit has tripped at what time.

CHAIRMAN KRESS: In principle, I think I would rather have ten 100-megawatt modules to deal with than one 1,000-watt module.

MEMBER ROSEN: You would?

CHAIRMAN KRESS: Yes. Because --

MEMBER ROSEN: Not I.

CHAIRMAN KRESS: Well, I think I would. In the first place, I've got a lot more data because I'm looking at each 100-megawatt. I've got a lot more information about each 100 megawatts. I've got a limited dependence of one on the other. There's very few common causes I think, maybe earthquakes, maybe even tornadoes. But I can't see how one module is going to affect another one very easily. And I've just got to -- I've subdivided my problem into smaller units that I can deal with.

MEMBER ROSEN: And I would say you've multiplied your problem by ten. Instead of having a three-ring circus, you've got a ten-ring circus.

CHAIRMAN KRESS: It depends on your viewpoint.

MEMBER ROSEN: You've got three of the units in Outage 7 and the other units running of which two are at ascent, two are at descent, the other three are at stable.

MEMBER SIEBER: What we did at Beaver Valley when we faced this problem was we built a seismic glass wall through the middle of the control room and kept Unit 1 operators on one side and Unit 2 operators on the other. And the only thing you could see from one unit to the other was which ones were sweating the most.

(Laughter.)

MR. FLACK: That makes them independent.

MR. ARNDT: Actually, one of the issues that has been raised by one of your former colleagues, Professor Miller, is to basically make a ten-unit plant look like, from an operational standpoint, a one-unit plant. So it really is a combination of both the issues that are discussed here. So it's a very complicated human factors I&C issue from an operational standpoint.

MEMBER ROSEN: At South Texas, there were two identical units but with two control rooms. The units are 500 yards apart for the purpose of so they don't confuse each other.

MEMBER SIEBER: That's right. That's important.

MEMBER ROSEN: It's important. And also when one unit is in shutdown and the other on is running, you can take some manpower from the shutdown unit to help the operating unit if it gets into trouble.

MEMBER SIEBER: Well, I exaggerate the problem because really what the shift manager has to do is exercise discipline over his crew to make them pay attention to their job. And in coal plants, that sometimes didn't happen. In the nuclear plants, the discipline's pretty high.

CHAIRMAN KRESS: Well, I think it's a manpower issue.

MR. FLACK: Yes. And that leads us to that second bullet there, staffing versus in light of these multi-modular designs and how much staff are you going to have to deal with these like normal?

MEMBER LEITCH: Even with two units you get -- operators get mixed up too and have gone to the wrong unit. With ten, I would imagine that would be much more complex.

MEMBER SIEBER: We solved that with colors, but I don't even have ten colors.

MEMBER LEITCH: Yes, we did that too, with color and striping on the units and the procedures were --

MEMBER SIEBER: We painted the walls and everything.

MEMBER LEITCH: -- color-coded to correspond with the unit. But I mean there's a lot of those tricks you can do, but in spite of all those things, there's still an element of confusion.

MEMBER ROSEN: There's also bar coding now where you swipe the procedure that you're using and then you swipe the component you're on, and if they don't -- if it doesn't agree, you're in the wrong unit or you're on the wrong component. So that's one issue.

But the other issue that I think you're alluding to is the incredible numbers of people they're talking about or the very few numbers or the very few people they're talking about operating these things, because people cost 70 percent of the total for an operating plant. So if you can get that down by an order of magnitude, you've knocked a big chunk of operating costs out. But I've heard numbers that are absolutely incredible in terms of how few people they're talking about having running these plants. Is that something you're going to look at, workload, task workloads and stuff like that?

MR. FLACK: As we learn more about what their plans are, we would certainly be looking into that. What is the role of the operator in these cases with multiple plants? And reliance on I&C to do most of the job. The one thing also is this third bullet, the time that you have. Now, clearly, in many cases, you have a lot of time to react so you can get people to the site, for example. But then on the downside is could the operator do something trying to help and does something that causes -- that compromises the situation, causes an adverse situation? So that's the flipside of that. So these are issues that would need to be prepared for to deal with when they come in.

MEMBER SIEBER: There is a piece of history. The plants that were built around the time

Surrey was built may -- in the design concept of the building layouts, they would build a locker room. In our plant, our first unit had 75 lockers so each person could have a locker that was employed at the plant. When I left there, there was 1,200 people, and we had buildings with locker. So people's first estimate when they sell a power plant to the utility execs is you aren't going to need -- this plant is fail-safe and it's totally automatic, and you aren't going to need people, and it just never works out that way.

MEMBER ROSEN: It turns out paper reactors are very easy to run. Require few operators.

(Laughter.)

MEMBER SIEBER: Not one has had an accident.

MR. FLACK: Okay. And, of course, the models that need to be -- to support the PRA they do come in with and the treatment of human reliability and within the context of those models is something that is going to be a challenge.

The next viewgraph is right along the same line we've been talking about. I&C and the application reliance of advanced I&C for process control and multiple modules. Again, it's the reliability issue, the failure modes an effects analysis, the systems interactions that could occur possibly amongst the modules and the I&C may present a problem, a challenge, and then, again, the models to support the PRA in light of all that.

So at this point in time they're mostly staying engaged with what's going on in outside world and thinking ahead, but there's not too much one can do without again, having a design in and seeing exactly what it is that they're going to rely on with respect to INC. I don't know if Steve Arnot is actually the author of that section of the record.

MEMBER ROSEN: Let me ask him a question. Are we talking about continuation of the IEEE 279 requirements for separation of church and state for the protection and control? Or is this the place where we the cross the rubicon in terms of that?

MR. ARNOT: There has been some discussion both in DOE research programs and in the vendor discussions --- much more highly integrated control systems for safety/non-safety, etcetera. It's integrated in the control room and integrated some of the balance of plant systems, integrated in the switch yard. So there's a lot of issues associated with both integration across safety/non-safety and also integration of non-safety balance of plant-type issues.

That much being said, no one has come in and said we would like an exemption from these rules or we would like to change it, etcetera, either 279 or 603 or anything like that.

One of the real issues is if you're going to have a framework that is more heavily structured on risk reliability type of standpoint, how do you deal with digital system safety and things like that? And we already have in place some research programs that are looking at that both in terms of things like isolation common loop failure and those kinds as issues as well as actual coming up with numbers for digital failures, which is a non-trivial area as you are aware.

The efforts we're doing in addition to that work for the advance reactor program is looking at some of these specific issues and how that affects the ongoing work we have in place, like multi-modular issues, like some of the more highly integrated systems like things like the trade offs currently. The isolation in other issues has driven the trade offs on diagnostics versus simplicity to the simplicity standpoint. Most of the digital system retrofits we're seeing are relatively simple digital systems.

When you go to whole new digital design, and this is the first time anyone in the United States has done a completely new digital design, you get people thinking about much more complicated systems, with failure type detection systems with online diagnostic systems, things like that that complicate the systems much more highly, integrate the systems much more highly, than you would logically ever put in a retrofit. So we're planning on looking at things like that that you would see in an advance reactor that you would not see on a retrofit. That's not really a complete answer to your question, but we just don't know at this point how far they're going to go down that path.

MEMBER ROSEN: Well, the owners will decide that I think. But to some extent we need to move forward I think with digital systems. We can't stay where we are. On the other hand, where we have been I recall hearing when Y2K came about, about how robust it was in the nuclear industry because we didn't have all these digital systems. We didn't have to worry about the fact that this date glitch was going to bite us because our systems just didn't know anything about that. It was really a sobering -- if you think about that for a little bit, it told you something about the value of analog systems.

Well, we can't go there anymore, but I think we should not lose sight of the value of some of these old concepts, the separation of control and protection circuitry, and somehow manage to bring across the boundary into the new world, some of those concepts that have served us well in the past. On the other hand, in the digital systems you have a whole lot of other things you talk about, online diagnostics and fault tolerance and multiple power supplies and a whole lot of things that are of real value.

MR. ARNOT: You also have a lot of potential cost saving things like multiplex systems where you don't have to run as much wire. You have fiber optics, you have wireless sensors. You have a lot of things that vendors would see as very cost effective, but also drive you towards some of these questions that are going to be real issues.

MEMBER ROSEN: I understand there's a value in cost, but I was more interested in some of the values in safety of the new equipment. New equipment could have a lot of significant advantages in the safety area including default tolerance, for example. Diagnostics, self diagnostics, systems that turn themselves off and announce they're turning themselves off and why, and transfer control to another operating system. So there's a lot to be said for these hardened systems.

MEMBER SIEBER: Well, the conversion of an operator from an analog to a digital system is sometimes difficult. For example, when the airlines changed from analog instruments to glass cockpits, there was a lot of upset pilots because they really liked the old stuff better. On the other hand, the younger folks like the new stuff and don't like the old stuff. So there is a sort of trial for some people when they make the conversion.

MR. FLACK: Okay, another area of the plan is structural analysis section, and this was authored by Syed Ali, who is with us, Harmon Graves, and to some extent, Joe Muscara. And this area deals with the integrity of the reactor vessel and the confinement of building and structures and dealing thing with seismic, so on. The technical issues in this area, and challenges are summarized in these five bullets. Concrete, and of course, concrete having to preform at higher temperatures and then how does it age under that environment. The applicability of current industry codes and standards, the modular HTGR designs and how they're constructed and mass produced, and what kind of inspections would have to occur under those conditions. Seismic response of connected vessels. We were talking about the connected inner connections of the pipe before, how these will respond under seismic condition. And as well as graphite structures, how they will be performing under seismic conditions. Soil structure interactors. We know the modular designs are going underground and how these will behave, also again under seismic events.

CHAIRMAN KRESS: When you talk about looking at underground effects, you don't mean the whole reactor is underground. You just mean that part of it is underground.

MR. FLACK: The GTMHR is in a silo, which is a deeply embedded structure which is level with the surface. Now the original PBMR was only, I think, two thirds underground. And I don't believe that was totally underground. But these are deeply embedded structures.

CHAIRMAN KRESS: That's what you mean by underground?

MR. FLACK: Yes, that's right. Not in a cave somewhere, but I mean it's in in a silo.

MEMBER ROSEN: An AP1000 would be a deep hole.

MR. FLACK: Again, the challenge is performing risk informed inspection and service inspections for these structures throughout their lifetime. Syed, I don't know if you wanted to add anything to that at all at this point?

MR. ALI: This is Syed Ali from the staff. Just back on the soil structure interaction, I just wanted to add that most of our review expedience for the existing reactors have been for structures that are maybe partially below ground, but mostly above ground. So under an seismic event, if the majority of the structure is underground, than some of the dynamic pressures, soil pressures acting against the structure are phenomena that are non-linear and not so well understood and so we need to further develop that experience. I think that, like you said, there maybe other cases where as far various reasons, at least for the future plans that might be more underground, more sheltered than they are.

MEMBER SIEBER: Yes, there are other effects that go on there too. Shipping Port was built underground with just small percentage of its reactor plant surface of above the ground. Some of the effects were that the concrete enclosure that it was in, subject to the groundwater pressures, okay, was put in over large areas. That can be a significant force which causes cracking and leaking and all kinds of things. There's more to it than just soil liquidity and external forces.

MEMBER ROSEN: Syed, wouldn't it be true to say that there's considerable amount of experience with seismic forces on underground structures?

MR. ALI: There is, for example, for tunnels and things like that. But the sophistication and the level of analysis that you do for nuclear power plants is much higher sophistication. There is some experience on the west coast, but even there there's a lot of difference between doing a detailed, dynamic time history analysis the way we do for the structure versus some of the codes that they use on the west coast, which are superstatic analysis for seismic effects.

Plus our staff does not have the experience because they have been involved in nuclear structures which have been traditionally above ground.

MR. FLACK: Okay, thank you Syed. And that leads us then to our last area, research area, consequence analysis and basically on this one we're looking for differences in chemical forms and radionuclides that might involve from these new plant designs as well as the timing of the release and what we would might or might not need to do to MACCS to treat these differences, both in the technology of the designs and in the biological factors that result from the different chemical forms, radionuclides that would be released.

And then there's the follow on discussion which is being entertained as a possible policy issue about the length between the consequence analysis and emergency planning, for example, and the size of the EPZ. So those are some of the technical issues and challenges we see with respect to our ability to do the consequence analysis for these event plans. And Jocelyn Mitchell is with us. I don't know, Jocelyn, if you wanted to add anything to that since you had that section of the plan. So, no further questions?

CHAIRMAN KRESS: On the issue of input into MACCS, of course, there's the timing and mix of isotopes and quantity of fission products, but usually there's an energy associated with -- you have to have an input for the plume, an energy input. Is that part of what you're looking at here also?

MR. FLACK: Well, I would think that MACCS would have to deal with that at one point and the period of time over which the release will take place, for example, which could be days instead of hours.

CHAIRMAN KRESS: Do you have some criteria, for example, for gas cool reactor you concluded you couldn't get any fission products released for x number of days, you wouldn't have to have any evacuation emergency planning, you could just ad hoc? Do you have criteria like that?

MR. FLACK: That's a question of whether the Commission wants to entertain such criteria at this point. We're in severe accident space.

MR. ELTAWILA: We are planning to address that as part of the policy issue that John mentioned which will be coming out this fall, you know, so that's one of the questions.

MEMBER SIEBER: That's more of a political question --

CHAIRMAN KRESS: Well, it's political, it's defense-in-depth, it's a lot of things.

MEMBER SIEBER: Yes, but if you have an accident some people are going to take off even if they're already 50 miles from the plant.

CHAIRMAN KRESS: They're going to have ad hoc evacuation then.

MEMBER ROSEN: It seems to me what the ACRS can add to the discussion is to try to focus on the technical issue.

CHAIRMAN KRESS: Like distributing chaos?

MEMBER SIEBER: I think that's where we should restrict ourselves.

MEMBER ROSEN: Yes, because the politics are the politics and we don't have much to say --

MEMBER SIEBER: I agree with that.

CHAIRMAN KRESS: We should always focus on the technical.

MR. FLACK: That leaves me with my final view graph if there are no other questions. And this is future actions. We discussed earlier this morning and again later this afternoon about the expansion of the plant to capture these new plants coming our way, specifically the ESBWR and ACR-700 and the SWR-1000.

CHAIRMAN KRESS: I understand the ACR people finally got smart and are going to cool with light water instead of heavy water.

MR. FLACK: That's my understanding.

MEMBER ROSEN: It's a light water and heavy water machine. The advantages of both and the disadvantages of both.

MR. FLACK: That's right. So there will

be competition for the funding as Farouk had mentioned earlier which will play out over the next several months. So it's important, I think, at this point also to consider that and any letter that the ACRS writes on the subject plan comprehends completeness of the plan as well as where the scope of the plan addresses now in light of these other plans coming in.

CHAIRMAN KRESS: Do the Canadians have a PRA for their Candu reactors?

MR. FLACK: That I don't know.

MR. ELTAWILA: Not yet, but they are aware of the need to provide a PRA.

MR. CARLSON: They did provide one with Candu 3.

CHAIRMAN KRESS: Yeah, I wondered.

MEMBER ROSEN: Jack, did the last bullet refer to ACRS Members?

MR. FLACK: The last one?

MEMBER ROSEN: Yes, the last bullet.

MR. FLACK: Implement and recurrent --

MEMBER ROSEN: Trying to stay alive through this?

(Laughter.)

MR. FLACK: I don't know about that.

CHAIRMAN KRESS: It's a living document.

MR. FLACK: It means that we would certainly be flexible in consideration of other activity.

MEMBER FORD: John, in terms of the first bullet, Farouk mentioned there might be extra funding coming. Is there not a preeminent limitation of manpower?

MR. ELTAWILA: There is none in the light water technology. I think we are able to identify expertise in-house here and outside to be able to help us in light water technology. Definitely, as you are aware, there is limitation in manpower in-house and externally in the gas cooled technology. ACR, you know, it's still, although it's a light water reactor, but it's a new concept to us, the horizontal core and pressure tube and so on. So we need to educate ourselves.

So as far as the ASPWR, I don't think we have any limitation in that regard.

CHAIRMAN KRESS: For this, a lot of this research you may end up doing all your on. It's not particularly required of the licensee or the applicant, will you direct funding from Congress for that? This won't come out of fees and charges to --

MR. ELTAWILA: No, most likely. That's

the problem the Commission faced that all the research funds would be be charged to the licensee so it makes the Commission, puts the Commission in an awkward position why all this utility would pay research for gas cool reactor. So I don't think, I don't know what the Commission is going to do about requesting that additional fund, but it does not look like separate from the fee based fund.

CHAIRMAN KRESS: This sure would be a good place to have it separate.

MR. FLACK: Okay, the only other thing I wanted to mention was that we will meeting with the ACNW later this month to talk about material safety and waste renewal and then ultimately transmit the plan to the Commission this fall along with the policy issue paper that Farouk mentioned earlier. And then this document would be maintained living and work being coordinated with the user offices and maintaining it that way.

MEMBER FORD: As you see it right now, John, the plan that you submit to the Commission, how different will it be from the one we have in our books right now? For instance, will it include items coming from PERT activities, privitalization activities?

MR. FLACK: No, I don't think we'll get

too much difference from -- these other activities, I think are something we have to think about, the first bullet and the new plans that are coming our way will certainly need to be captured within the plan as best we can and transmitted to the Commission. The fact that either these light water reactors and that we're better prepared to deal with them wouldn't expect too many technology gaps that we might say that we need to fill and maintain for the long term as we do with the HTGRs, for example.

So I'm not envisioning any major differences too much with the way the plan is written now. A lot of the, I think, as we transmit the plan to the Commission, we certainly need to discuss how we plan to carry out and implement this plan over the long term and we will maintain it. And I think that will go in the SECY itself as we transmit it to the Commission. But as far as the plan is concerned, I don't see major changes to the plan from now until then.

MEMBER FORD: Okay, the reason why I asked the question is you know we committed to the Commission the research report that we have to write will be on advanced reactors. So this will be the material that we will be basing the report on. Is that a fair comment?

MR. FLACK: I think that's fair. I think we just about drained everybody we could.

MEMBER SIEBER: Just send this in.

CHAIRMAN KRESS: Yeah, just put a cover letter on it.

(Laughter.)

MEMBER ROSEN: One of the things we talked about this morning was that you had acknowledged a need to put more in it about a view of what's going on in J4.

MEMBER FORD: Will that be included in this?

MR. FLACK: Well, I think it would be more of a status of what is going on outside the group this plan originally centered on for and expand it slightly to capture these, but to recognize these other designs that are going on. Now we could incorporate that as an appendix that continuously gets updated as we get more information. I don't think there will be too much of an impact of that on the actual activities as we see them today since these are conceptual in nature and we need to follow them closely to see if there are needs, issues as they arise. But within the next few months, I don't see a major change to the plan.

MEMBER ROSEN: I just don't think you would be serving the Commission well or the public well if you didn't acknowledge all this other action going on in the world and acknowledge that, although it might not have an impact on next year's plan, it will surely have impact on the out year plans.

MEMBER FORD: Will there be any comment at all on the NEI document that's just come out?

MR. FLACK: At this point --

MEMBER FORD: Stakeholder interactions and I'm wondering if that would include that.

MR. ELTAWILA: Taking about the framework? That's already been acknowledged in the risk inform regulatory implementation plan that we sent an update to the Commission this past June and acknowledge the NEI paper and it tried to relate the NEI paper to the existing risk inform regulation and what we are planning to do for advance reactor. So it is in the EDO and once it's signed it will be available.

I'm sure that the DRA came and discussed this with you all before it went to the Commission. Or at least I hope so.

MEMBER LEITCH: In the description of the PERT process that begins on page 109 of the report, there's a six step process outlined which really describes the PERT process. I don't see clearly in that description an assessment of the viability of a particular type of reactor. Is that thought included in there?

MR. FLACK: That is generally not included as part of the PERT process when a PERT focuses on a particular technical area. I don't know, Don, do you want to comment on that?

Generally it's within a certain context. If it's HTGR, it would be focusing on fossil fuel behavior and so on.

MEMBER LEITCH: What I'm saying in assigning priorities, where does the differentiation between the likelihood of building type a verses type b verses type c. How does that enter into the prioritization process?

MR. CARLSON: I don't think that comes under PERT per se, that comes in at a different level. I think Farouk alluded to that on the seriousness of an application.

MEMBER SIEBER: If somebody sends in an application, you have to deal with that application. It's their decision and their move.

MR. FLACK: Basically you do it through pre-application.

MEMBER SIEBER: It's the way it works and you keep raising issues until they're either successful or give up.

MR. FLACK: And we saw that shift just happen with the pebble bed and now with the GTMHR, so now pebble bed has taken a back seat and GTMHR is the one we're looking closely at. So it's really, you know, a timing issue on the part of who the applicant is and when do they want to submit design certification or a licensing application.

MEMBER LEITCH: So you don't really have a good handle on the viability of a particular project, that is at that stage? In other words, are we spending our scarce dollars where we are likely to get the most payback? That's a judgmental call that we haven't really made.

MR. ELTAWILA: That's a hard question and I think the Commission deal with this issue continuously about where they are going to put these resources. And again, we will come down a Commission policy, that we are going to be working on this application. I think the Commission, anybody submit application to us we will have to consider that. And again, for other means, for example, most applications that will have more serious consideration at NRC are those are the ones that will be supported by utility which when you have utility come in concert with a vendor and say we would like to decertify this design, that will add more credibility than you have a vendor that just want to get the certification for design.

And we take that into our budget process. Not in the PERT process. The PERT process, as John and Don indicated, focus on the technical issue and where you spend your money on getting efficient product release model or on getting high temperature material or something like that.

The budget process is the one that's going to take into consideration the seriousness of the application, the support from the industry behind that application.

MR. FLACK: I should also mention that the plan itself, there are activities of the plan that are currently ongoing. It's not that we plan to do everything that's here. In fact, some of the work that's in this document is work that's going on. The question becomes which priorities and how do you prioritize future work? There's a certain level of work that needs to be maintained, for example, in graphite. A year ago, we had no one that was an expert on graphite really in the Agency. And now we are developing a person with those kinds of skills -- Shreeni was here earlier.

So we're actually doing some of this right now. And there's a certain level that one might have to say that infrastructure should be at a certain minimum, it should have a certain minimum expertise. And that would sort of take the highest priority so you'd be able to at least ask the right questions.

And then the question is is when you exercise this infrastructure, what are the activities then that you will do? And that begins, well how do we allocate our resources to do those activities? So it's like another level.

But there is this minimal level that I think the Agency needs to maintain if we're serious about gas cooled designs. And that would be an expert on all kinds of fuels to stay tuned in that area with what's going on internationally, participation with the DOE projects and so on. And things like graphite where we have somewhere here that can stay involved and engaged in that field. So when we prioritize that we don't eliminate those positions and say, well we don't need them right now. We'll go and get them later on. So I really believe there's some level we need to maintain.

MEMBER LEITCH: I had another question in the area of, the rather large area of fuels and materials. There was virtually no discussion of research activities for advanced light water reactors. Is that an issue of prioritization and some of that has been screened out? Or we just don't believe there are significant issues in fuels and materials for advanced light water reactors?

MR. CARLSON: No, the fuel section of the research plan did have a discussion of IRIS.

MEMBER LEITCH: COLLINS: Yeah, right.

MEMBER SIEBER: That has significantly different characteristics in the other light water content. I presume that the fuels in AP-600, AP-1000, BWR are pretty much the same as the concepts in current generation.

MEMBER RANSOM: Since the plan is focusing on gaps, changes, differences between now and the future.

MEMBER LEITCH: So the absence, for example, of discussion of that in the materials section, the discussion of advanced light water reactors is not, some of that has been screened out for budgetary reasons or priority reasons, but just that no significant gaps have been identified.

MR. FLACK: That's right. We are doing that work as we speak, so it wasn't trying to capture all the research we do. It's really to try to capture those gaps that we see.

CHAIRMAN KRESS: Let me ask you a technical question. Somewhere in the document I read that you need to look at critical flow at much lower pressures because the reactor depressurization, I guess it was AP-1000, I'm not even sure of that now. Could you explain what that means to me?

MR. FLACK: Critical flow?

CHAIRMAN KRESS: No, I know what critical flow is. I don't know why you're now saying it's going to occur at much lower pressures. Is that because the reactor depressurization does not take place isontropically as opposed to slow depressurization? See, I don't understand why slow depressurization and rapid depressurization gives you a lower pressure for the critical flow.

MR. FLACK: I could speculate. That could be dangerous.

MR. ELTAWILA: How about if I get back to you? I know Richard mentioned that --

CHAIRMAN KRESS: The only thing I could suspect was the rapid depressurization might not be isontropic.

MR. LEE: Yes, I think the pressure is also lower. You can get it down there faster. The data base we have, we believe that mostly in the high pressure region or the critical flow. And then the feedback from the containment also affects the flow itself. So looking at those two in combination. But it's not a critical area that will stop the AP-1000 certification. It's just completeness for the database. Off the record, I'll tell you the other reasons.

(Laughter.)

CHAIRMAN KRESS: Okay, I appreciate it. I thank you very much. How are you going to condense this into an hour and a half?

George is going to be interested in the framework. But you need to have some words there, not the full thing, but a few words. Dana is going to be interested particularly in fuels and everything else also. Bill Shack is going to be interested in materials issues and everything else. So those are the things that we want to get across to the missing members.

MEMBER SIEBER: I think in the issue of the framework, I think that's really important. And it seems like somehow or another it's not getting the attention I think it needs. So maybe talking about it again so everybody understands how important it really is.

MEMBER ROSEN: We tried to probe that this morning a little bit. But how do you decide what's a design basis accident and what's not? Or alternatively, the model of proof offered which is you don't try to decide. You just leave that aside and just say we're going to talk about risk and risk analysis and have a continuum of spectrum. I think that whole discussion, George is going to be very interested and Dana will too.

CHAIRMAN KRESS: Yeah, I think Bill will too.

MEMBER SIEBER: I agree with you, Steve. I think it still needs more working out. There is a pretty slick way to do it, I think. You know without sort of riding the line between deterministic and probabilistic analysis. And I would prefer the Agency set the tone as to how the regulation should be than have an industry group or somebody else come in and do that.

MEMBER ROSEN: Well, I think there is some good ways to do it as you suggest. But I also think we're entering into an area where there's a lot more uncertainty than there had been in the past. So I'm unlikely to say such things. The defense-in-depth margin is something at the outset turns out, you know, is crucial.

MEMBER SIEBER: You could put it on with a rational basis as based on PRA or you could put it on a deterministic basis because it feels good. And I'd rather be more --

MEMBER ROSEN: We're going to have a lot of uncertainties. We've heard about them, a lot of them, today. And so I think the discussion of how the uncertainty is dealt with with new technology and what we've been raising here is going to be of central interest to the three remaining Members who aren't here. Eight of us are here.

MR. FLACK: Okay. But although framework is only one piece of that bigger plan, there's a lot of the plan and I think it would be a disservice for me to try to summarize that plan in the short period of time. I mean, I can identify the different areas and maybe touch upon a couple. It would be tough to try to go into each subject and try to summarize each subject in an hour and a half. Plus the framework. That would be quite a challenge.

MR. ELTAWILA: If I heard, I think we need to have a presentation that covers the framework, if you will, and the materials.

MR. FLACK: You want to do that?

MR. ELTAWILA: I think we will have to do that.

MEMBER ROSEN: Well, I think some of the discussion on the neutronics was also quite useful and you can't, unless you're going to cover that in the fuel, I think you have to mention something about reactor systems analysis.

MR. FLACK: Then thermal hydraulics.

(Laughter.)

CHAIRMAN KRESS: How fast can you talk?

MEMBER ROSEN: You can talk as fast as you'd like, but you're not going to get more than about four words out before --

MR. FLACK: I think I got four vu-graphs the last time. I think that was it. It was over at that point.

MR. ELTAWILA: I will be about ten minutes each topic.

MEMBER SIEBER: Maybe the way to do it is instead of going into such great detail about what each one of these things is, is to come up with a list and say these are the projects and a one liner as to what it is you're trying to do and why it's a gap and how you're going to fill it.

CHAIRMAN KRESS: That requires making new vu-graphs between now, and I don't that's --

MEMBER SIEBER: Between now and Thursday?

CHAIRMAN KRESS: Yeah, I don't think they want to do that. I think I would select from the vu-graphs you have some way and --

MR. FLACK: Well, I could attempt to do that. I mean, we have 26 people working on the plan so I get all --

CHAIRMAN KRESS: That's up to you how you want to do it.

MR. FLACK: I can have backups and try and do that.

MEMBER SIEBER: Well, I don't we ought to make you do more work than necessary. I agree with you.

CHAIRMAN KRESS: That's one drawback with having the Subcommittee this close to the full.

MEMBER SIEBER: One thing you could do is just take the table of contents which is right near the front of the plan and make a vu-graph out of that. And that tells everybody what's in it.

CHAIRMAN KRESS: But you know, personally, I think you can probably assume that these three Members have read this. They're generally pretty good --

MR. ELTAWILA: I know Dana.

CHAIRMAN KRESS: Dana, you can be sure. George may not have had time to do it all.

MEMBER SIEBER: But he will do his part.

CHAIRMAN KRESS: He'll do his part. And Bill usually reads the things, too. You know, they won't come in not knowing anything.

MEMBER SIEBER: Yeah, they won't come in cold.

MEMBER ROSEN: You've dealt with the easy ones here.

(Laughter.)

MEMBER SIEBER: Yeah, we argue with each other.

MR. FLACK: That's why we finished on time.

CHAIRMAN KRESS: Well, you know enough now to figure out how to --

(Laughter.)

MR. FLACK: We'll put something together.

MEMBER SIEBER: I guess if I could offer

a general statement. I thought the plan was very comprehensible. Well put together. Well done.

CHAIRMAN KRESS: I agree. It was a very nice piece of work. Well done. I am going to keep it as resource document because it's got the issues in there and what people are doing at various places. I thought it was very nice.

MEMBER SIEBER: I guess the other thing that concerned me was the same thing was concerning Graham Leitch is that you've got a limited pot of money and a limited amount of resources and you've got to sort of guess which concept is going to be the hot concept of the day so that you aren't spending money on something that will never be built.

CHAIRMAN KRESS: I think they always have to have to have, they're always faced with that problem. They know how to do that.

MEMBER SIEBER: But I don't, so.

(Laughter.)

CHAIRMAN KRESS: We'll leave that up to Farouk. He knows how to do that. Well, I appreciate these very nice, very good presentations.

MR. FLACK: Thank you.

CHAIRMAN KRESS: Good work. We'll look forward to see how you can --

MR. FLACK: That we can stay below that hour and a half?

CHAIRMAN KRESS: With that I'm going to declare this Subcommittee meeting adjourned.

(Whereupon, at 5:38 p.m., the meeting was concluded.)

## CERTIFICATE

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

Name of Proceeding: Advisory Committee on Reactor Safeguards

Docket Number: N/A

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.

/S/ Debra Wilensky

Debra Wilensky
Official Reporter
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