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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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SUBCOMMITTEE ON FUTURE PLANT DESIGNS

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TUESDAY, APRIL 5, 2011

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 10:00 a.m., Dr. Dennis Bley, Chairman, presiding.

SUBCOMMITTEE MEMBERS PRESENT:

DENNIS C. BLEY, Chair

SAID ABDEL-KHALIK

J. SAM ARMIJO

CHARLES H. BROWN

MICHAEL CORRADINI

HAROLD B. RAY

JOY REMPE

MICHAEL T. RYAN

CONSULTANTS TO THE SUBCOMMITTEE PRESENT:

THOMAS S. KRESS

NRC STAFF PRESENT:

MAITRI BANERJEE, Designated Federal Official

SUDHAMAY BASU

KATHY GIBSON

STEPHEN FLEGER

JOSEPH KELLY

MOURAD AISSA

HOSSEIN ESMAILI

STUART RUBIN

MAKUTESWARA SRINIVASAN

SHAH MALIK

AMY HULL

JOSE PIRES

JEFFREY WOOD

MARY DROUIN

SEAN PETERS

YAGUANG YANG

RUSS SYDNOR

ALSO PRESENT:

DAVE PETTI

T-A-B-L-E O-F C-O-N-T-E-N-T-S

P-R-O-C-E-E-D-I-N-G-S

10:02 a.m.

CHAIRMAN BLEY: The meeting will now come to order. I'm Dennis Bley, chairman of the Future Plant Design Subcommittee.

We have with us today ACRS members - I've got to recheck my list - Harold Ray, Sam Armijo, Joy Rempe. We expect very shortly to have Dr. Abdel-Khalik and Dr. Corradini join us. And there may be some others.

Tom Kress is here who is our consultant. Maitri Banerjee of the ACRS staff is the designated federal official for this meeting.

The purpose of today's meeting is to receive a briefing and discuss with the staff the NRC High-Temperature Gas-Cooled Reactor Research Plan that addresses work needed for the NRC to prepare to review the future NGNP Application.

The last time the subcommittee had a briefing on the Research Plan was on January 14 and 15, 2009. That was quite some time ago.

The Research Plan has been updated to address only the HTGRs and the discussion on sodium-cooled fast reactors has been eliminated.

Drs. Corradini and Rempe may have some

potential organizational conflict. Hence, they will not take part in any discussions specifically related to their work.

The rules for participation in today's meeting were announced in the Federal Register on March 22nd, 2011, for an open-closed meeting.

I'm asking the NRC staff and the applicant to identify the need for closing the meeting before we enter into discussion regarding any proprietary material, and to verify only people with the required clearance and need to know are present.

We have a telephone bridge line for the public and stakeholders to hear the deliberations. This line will not carry any signal from this end during the closed portion of the meeting.

Also, to minimize disturbance, the line will be kept in the listen-in-only mode until the last 15 minutes of the meeting. At that time, we will provide an opportunity for any member of the public attending the meeting in person or through the bridge line to make a statement or provide comments.

As a transcript of the meeting is being kept, we request that participants in this meeting use microphones located throughout the meeting room when addressing the subcommittee.

Participants should first identify themselves and speak with sufficient clarity and volume that they can be readily heard.

We will now proceed with the meeting, and I call upon Dr. Sud Basu of the Office of Nuclear Regulatory Research to begin the staff presentations.

Sud.

MR. BASU: Thank you, Mr. Chairman, and good morning ACRS members. We are waiting for - we are waiting for Ms. Kathy Gibson - oh, there she is.

Kathy, would you like to say a few words before I start?

CHAIRMAN BLEY: Thank you. Welcome.

MR. BASU: Kathy is the director of the Division of Systems Analysis and we have the cognizance of the HTGR R&D program for the Office of Research.

CHAIRMAN BLEY: Okay. Kathy.

MS. GIBSON: Hi. I'm just going to take a minute and introduce myself. I have replaced Jennifer Eule who is now the Office of Research deputy director.

And my name is Kathy Gibson. I was Jennifer's deputy for two years. So, I'm not entirely new to the work that we're doing here.

And I just want to say we welcome the opportunity to present our HTGR research plan. It's been many years in the making. Fits and starts. We're still interested to see where this program is going to go for the long term, as I'm sure you are.

But the staff has put together a good presentation. So, we're interested in your feedback. Thank you.

CHAIRMAN BLEY: Thank you.

MR. BASU: Thank you, Kathy.

As Dr. Bley pointed out, we had the last briefing before ACRS a little over two years ago. We made a couple of attempts to come before you and brief you on the status of the program. For one reason or another, that did not happen.

We are glad to be here finally. And the good thing is that we will be able to share with you the progress made in the last couple of years.

So, I'm going to - let's see. How does this go? I'm not doing something right.

CHAIRMAN BLEY: Help is on its way.

(Off-record discussion.)

MR. BASU: Okay. So, you know, I'm here with a number of colleagues from the Office of Research. And between us, we are going to cover all the topics that are in the R&D plan that you have a copy now.

There is going to be altogether four new presentations, including my overview presentation. In the interest of time, I will request the Chair to time each speaker so that we can give you the full coverage and still be in time.

CHAIRMAN BLEY: And the Committee will try to allow you to proceed.

MR. BASU: Thank you. So, the outline of my presentation, which is an overview presentation which is going to set the tone for the rest of the presentations, is to talk about the objectives of this briefing.

I'll say a few words on the role and scope of NRC research for HTGR, high-temperature gas-cooled reactor, assumptions we made in developing the plan, as well as implementing the plan.

I'll spend a few minutes on implementation status, and that's where I'm going to talk about the problems over the last couple of years.

My presentation will still be fairly broad in the overview mode, but the subsequent presentations are going to cover individual topical area in far more detail.

And then I'll say a few words about going forward from this point onward.

CHAIRMAN BLEY: Okay.

MR. BASU: So, the objectives of the presentation I'll be briefing is to provide you an update, as I mentioned before, on not only the research plan, but also its implementation status.

We are going to solicit your feedback, your input, comments, etcetera, as we always do for any briefing.

We're also going to request a letter from ACRS and that of course after we brief the full Committee. And currently, my understanding is it's scheduled sometime in May.

Okay. So, in terms of the role of NRC HTGR research, we are focusing on development of analytical tools and capabilities to perform confirmatory safety analysis as we do for any other reactor types.

So, this is not unique as such to HTGR, but the idea is that we have to be ready to be able to provide technical support for licensing review. And also provide technical basis for any regulatory decisions.

The role of the research is also to

develop technical basis for identifying and resolving safety issues. These are - these could be technical issues and also policy issues, but providing the technical basis for making any resolution of these policy issues. And, also, technical basis for developing regulations and guidance.

The third role is unique to HTGR, which is development of staff technical expertise and review capabilities.

As you all know, that the Agency has been involved in the Light Water Reactor business, you know, for all the time at least that I'm with the Agency, and even before that.

We have had some brief past experience with the HTGR activities. Review, sort of, activities. The resources, and I'm talking about human resources in this case, who were at the time at NRC, there are very few and far between to find them.

So, one objective on role of the research is to develop staff technical expertise and review capabilities. And that's kind of unique to this.

MEMBER RAY: In this historical context, could you just touch on Fort St. Vrain, or will you, in terms of the NRC's involvement in it, if at all, or how it was involved?

MR. BASU: Well, first of all, it's long before my time at NRC. And at that time -

MEMBER RAY: Before my time also.

(Laughter.)

MR. BASU: If I am not misspeaking, some of the activities were carried out under the banner of AEC in those days. And then later on the NRC was created.

So, and of course we - I believe Fort St. Vrain was licensed under a different set of rules.

MEMBER RAY: Well, that's what I'm asking. If you're not prepared to speak to it, that's okay. I just would like to - you're talking historically. And, certainly, Fort St. Vrain existed and operated.

MR. BASU: Right.

MEMBER RAY: I'm just trying to make a connection there.

MR. BASU: Could we get back to you on that?

MEMBER RAY: Sure.

MR. BASU: Thank you.

So, and of course I have to acknowledge that there are some staff from the Office of New Reactors, which is our sort of regulatory counterpart, if you will. The program office. So, if anyone from

Office of New Reactors would care to address this at some later point, feel free, please.

The scope of HTGR research I have kind of listed here. Again, the major focus is on the development of confirmatory safety analysis tools.

We have codes, computer codes, evaluation models, models of physical phenomena, data, data for assessment, validation given for model development and improvement.

A number of major technical areas where we are focused in, these are listed here. Thermal-fluids, nuclear analysis, accident analysis.

And I'm going to introduce the staff members at this point who are going to speak to these topical areas.

The thermal-fluids, nuclear analysis and accident analysis are going to be covered by a team led by Joe Kelly, sitting in the back, who is going to talk about the evaluation model development. Also, code development. In particular, the PARCS/AGREE code. That's the coupled neutronics/thermal hydraulics code. And, also, supporting experimental programs that we have.

He will be followed by Dr. Mourad Aissa who is going to talk about the nuclear analysis development.

And then followed by Dr. Hossein Esmaili who is going to talk about the MELCOR development and assessment work.

We also are heavily focused in fuel and fission product transport, release and transport program.

And Stu Rubin, who is the senior level advisor for the Office of Research in our division, is going to give presentations on the fuel.

That will be followed by two presentations. One on graphite to be given by Dr. Srinivasan - Makuteswara Srinivasan. I hope I pronounced it correctly.

And then on the high-temperature metallic materials, that will be given by Dr. Shah Malik. And also a portion of that will be given by Dr. Amy Hull.

We are going to talk to you about structural integrity of systems and components, the structural analysis area. That presentation is going to be given by Dr. Jose Pires from the Division of Engineering in the Office of Research.

The work has been done by a number of staff members and their contact organizations. And the staff members are present here to answer any questions you have.

I will come back and talk to you about the process heat utilization. I understand that is the very last item in the agenda.

So, depending on how we proceed, you will probably hear very little or a lot from me at the time.

There are other technical areas that we're going to give presentations on. Mainly, probabilistic risk analysis. This presentation will be given by Jeff Wood from the Division of Risk Analysis in the Office of Research. Also, assisted by Mary Drouin.

There will be a presentation on the human factors given by Mr. Stephen Fleger from the Division of Risk Analysis again.

And the final presentation on the instrumentation and control technology to be given by Dr. Yaguang Yang. Assisted by Russ Sydnor from the Division of Engineering.

CHAIRMAN BLEY: Sud, if I remember our discussions correctly, the reason these are so short is because these are just going about through the planning stages now. You haven't done much work in -

MR. BASU: Some are. Some we barely started work. And, yes, you are going to hear about some work that was initiated, but more on the planning stage.

CHAIRMAN BLEY: Okay.

MR. BASU: That's correct.

There are some other areas that I am not going to cover, we are not going to cover in this briefing. And they are also not covered in the R&D plan. These are important areas.

We didn't cover them in the plan for various reasons, and I'm going to list them. I don't have a slide.

The fuel cycle is not covered in the R&D plan with the exception of some neutronics aspect of the back-end of the fuel cycle, spin fuel, et cetera, that Mourad is going to talk briefly about.

We have not addressed security and safeguards in an integrative manner in the R&D plan. We have not addressed siting in the R&D plan, and I'm not going to cover any of that.

And I'll tell you the reason why historically they were not covered. Now, remember in the Energy Policy Act, the NGNP was defined as a prototype plan to be constructed at Idaho in the DOE property.

DOE was going to assume the ownership of fuel, and of course the siting was that DOE protect the siting. And DOE was also going to take the ownership of the security aspect.

So, all this is combined for NGNP. And I should probably mention for the benefit of the audience here, NGNP, Next-Generation Nuclear Plant, is a high-temperature gas-cooled reactor plant. That's how it is defined in the licensing strategy that we submitted to Congress back in 2008.

So, these were not part of the licensing strategy, development. And as a result, these did not become part of the R&D plan.

These are important aspects. We have to revisit them as the time progresses. And at some point, we will probably have to revise our R&D plan with the appropriate R&D activities.

MEMBER RAY: Dennis, looking all the way ahead to the letter, which of course is after the full committee meeting, I think these carve-outs here will be something we'll want to take note of because things have moved on to the point where they are more important now than they were when this plan was developed, and that will need to be acknowledged.

CHAIRMAN BLEY: Thanks, Harold.

MR. BASU: Yes, I think you are absolutely right. The climate has changed. It is changing. So, we'll have to revisit these issues at some point.

I'm not going to talk about policy issues in this briefing. These - it is the parlay of the Office of New Reactors, and they have actually come before you to talk about at least one White Paper, one policy issue in the past.

And I understand there's a plan for them to come and brief you. So, I'm going to skip those and the associated White Paper review process.

We made a number of assumptions going forward developing the R&D plan, as well as implementing the plan, and these are listed here.

The first one is that out scope is, in large part, generic. And by that, I mean applicable to the HTGR technologies that are in the market currently. Namely, the pebble-bed technology, as well as the prismatic technology.

So, we try very consciously to design our research program to address as much as possible both technologies.

Again, at some point when the point selection is made, we probably have to revisit the research program and put our focus on the right technology. But for now, it's mostly generic in nature.

We heavily rely on the data from DOE-sponsored programs in a number of areas like fuel, graphite, high-temperature materials and so on and so forth.

And you're going to hear in the subsequent presentations, some of these subject areas. And particularly in the fuel area, we'll touch upon DOE programs in some detail.

We of course are going to rely on applicant-furnished data for plant-specific licensing review. And this is no different from any other licensing activity.

We do rely on the applicant-furnished data to make their safety case. We take the data and we conform through our confirmatory safety analysis activities.

MEMBER ARMIJO: Dr. Basu, do you have an idea of who the applicant will be for this first prototype?

Will it be a DOE as an entity or will it be some commercial entity, private or what?

MR. BASU: My short answer is, no, I do not have an idea at this point.

MEMBER ARMIJO: All right. You just don't know.

MR. BASU: I don't know.

MEMBER ARMIJO: Okay.

MR. BASU: We are going to rely on the availability of complementary data from international HTGR R&D programs. There are a number of programs. HTTR program, High-Temperature Engineering Test Reactor program in Japan. HTR-10, the High-Temperature Reactor program in China. There was a program in South Africa, PBMR. That's kind of cancelled at this point.

But there's also a program in Europe, Europe, EUROPAIRS program, which is a fall under RAPHAEL program.

So, we will benefit from data that's been generated, that will be generated on this program. And so, we'll rely on the availability of the data from these programs.

And then of course we'll rely on the national and international codes and standards activities. Primarily, of course, national. But in some specific cases, we'll sort of make use of the international codes and standards data.

The final bullet I have is - and I'm going to just leave it at that, is adequate resource allocation.

In the Licensing Strategy Report, we stipulated that we are going to be ready by 2013, primarily, you know, mostly ready by 2013 with our tools to be able to entertain a License Application at that time.

Again, the landscape is changing. We obviously don't know whether we are going to receive an application. 2013, it's probably relatively a safe assumption to say it's going to slide beyond 2013.

But, again, the R&D plan was predicated on our being ready, in large part, by 2013. And so, any resource implication thereby so that we can be ready.

And now that the professor walks in, we all have to rise, right?

(Laughter.)

MEMBER CORRADINI: I thought I could sneak in quietly.

MR. BASU: Okay. So, I think you expressed an interest to know what are the new developments since the last briefing.

Well, cosmetically there is one new thing. You saw last time a probably 270 plus pages of the Adverse Reactor Research Program document. We pared it down to about fifty pages or so.

We obviously have excluded a few things from there, and I'll just quickly run by those few things.

One is that we had a small section on the LMR in that ARRP, which is not included in this NGNP R&D plan that you have. It is strictly HTGR R&D plan.

The ARRP also had the - if you could picture the universe of research that would be conducted not only by NRC, but by applicants, by DOE, in this particular case, DOE Lab, the international community, what we did, we extracted and focused on the NRC research that we are going to conduct, and we are conducting of course, again, relying on data that will come from all the other programs. So, that's the cosmetic change that you have probably noticed already.

There are a few other things that I will just sort of go quickly through. There's some development that affect our R&D program.

As you know, there is a trend to go into a lower reactor outlet temperature for the HTGR design. And with that lower reactor outlet temperature comes the consideration of the choice of materials.

Now, if we go to temperature like 700,

750, we have materials that have been qualified, code qualified to that temperature, which we can then make use of in large part. There may be some residual research that will be needed.

So, our focus currently is on the lower reactor outlet temperature meaning that we will validate our tools with the existing data as a large part.

The tools themselves are not going to be different for - hopefully not different for even higher-temperature applications, but we will not have the means to validate these tools in any short term. So, that's a change.

You also probably heard about reconsideration of steam cycle for power conversion. It started out with steam cycle. At some point, it then migrated to direct cycle-bred, and now it's coming back to steam cycle.

With that, there's this issue of moisture ingress that is coming into the picture. That is sort of becoming a more important consideration. Something that we recognized in the past, but we left out from further consideration, because we thought it was going to be a direct thermal cycle. So, you're going to see some future activities in that direction.

And the final one I said, broadening scope of process heat utilization. Initially under the Energy Policy Act, the focus was on hydrogen co-generation, in large part, hydrogen co-generation for which to be more efficient, you need a higher reactor outlet temperature.

Now, with the landscape changing to lower temperature, you can actually use the process here for many, many more applications in just as efficient manner as you would for hydrogen co-generation at a much higher outlet temperature.

So, there is some broadening of scope which has, again, some implications on materials R&D, as well as the degradation of components in the process heat part of the equation.

There are some other developments with potential R&D, design or regulatory impact. I'm not going to go through in any detail with these.

Co-location at industrial sites, again, that's the changing landscape. Multi-module design, that comes with the commercialization of the plant and, you know, sort of economics is driving that.

The two items that I have, the last two items, consideration of fuel form. By that I mean the kernel in the TRISO particle fuel.

You have the TRISO uranium oxycarbide or the uranium dioxide. The fuel that had been used in the past primarily in the AVR and THTR and in the European reactors are uranium dioxide fuel.

So, we have experience in that fuel, but the AGR program at Idaho has focused on uranium oxycarbide fuel and fuel development. You're going to hear more about that.

Depending on where we end up with, the choice between uranium oxycarbide or with uranium dioxide, there may be some issue that we are going to look into it at some point.

And, finally, consideration of technology alternatives, I told you about the PBR versus PMR. We are not there yet in terms of the point design, point selection of the design. And when that comes, then we'll have to revisit our R&D program accordingly.

This is the only cartoon I'm going to show you only to make the point that I already made which is that so far we are trying to stay technology-neutral, if you will, generic as much as possible. But at some point, we'll have to revisit our plan and program.

So, with that I'm going to move on to the implementation status. I don't want to go through each and every line here in detail. There's some points that I want to make.

Thermal-fluids, nuclear analysis, accident analysis, that is our major focus area. That's where we are going to develop - we are developing the tools, confirmatory safety analysis tools.

And in that vein, there are two areas that we are concentrating on. One is the analytical development of the codes, both system level and more detail - more mechanistic, if you will, code, and also the assessment database.

We are also involved in a number of supporting experimental programs. These experimental programs are in collaboration with DOE under the DOE-NRC Interagency Agreement.

And that, by the way, is going well. Both parties are benefitting from that agreement. We have some experimental programs in place that are funded under that agreement. We are also benefitting from very close collaboration with the DOE main laboratory, the Idaho National Laboratory.

MEMBER ARMIJO: For those particular programs, is it co-funded by NRC and DOE, or is it funded primarily by DOE and NRC monitoring the work?

MR. BASU: Well, let me talk about the

program very briefly. We started that program under NRC funding, and now it is funded under the DOE-NRC Interagency Agreement program. So it's, you can say, fully funded under that agreement, but we started that program under the NRC funding.

There's a program that we are currently discussing negotiating and trying to put in place as the OECD-HTTR Loss of Forced Circulation. This is in Japan. And NRC will be, if everything goes well, NRC will be a participant in that program very shortly.

Then there are a number of programs in the university, and these programs are complementing the DOE programs at universities under the Nuclear Energy University Program initiative.

In the fuel performance and fission products area, our work activities are primarily analytical in nature. We work with very closely monitoring the DOE program at Idaho National Lab, the AGR program that you're going to hear more about it in the subsequent presentations.

In the graphite high-temperature materials, likewise our programs are primarily analytical.

We have listed initiative as more experimental program on graphite at the Oak Ridge

National Laboratory looking at the stored energy aspect of graphite. And, again, you're going to hear more detail about that later.

In the structural analysis program, we are concentrating - again, this is all analytical. We are concentrating on the assessment of concrete behavior at high temperature and under irradiation environment.

We're also looking at the seismic and soil-structure interaction, and seismic loading consideration. I'll leave it at that.

Digital I&C and human factors there, we are looking at the instrumentation and control area. Our work at the moment is more like literature survey/review of what is there, what is out there, what we can benefit from in order for us to be prepared for any sort of new design that's coming in for -

CHAIRMAN BLEY: Is there anything we're seeing there that's different from what we're seeing in the new Light Water Reactor designs that are coming through?

MR. BASU: I think I'm going to defer that to -

CHAIRMAN BLEY: This afternoon. Okay.

MR. BASU: -- this afternoon.

CHAIRMAN BLEY: Fine.

MR. BASU: I would tend to say not much, but I may be totally off base on that.

In the human factors area, again, the focus is to update the guidance documents based on of course the information that we are going to gather.

In the probabilistic risk assessment area, we have conducted a planning study to identify the PRA needs and scope. And so based on that, we are going to initiate some future work on this.

So, going forward we'll continue to focus on the R&D that is, as I said, generic as much as possible. But at some point, there will be a bifurcation and we'll focus on the right technology.

We'll continue tracking the DOE NGNP program and modify our R&D recent activities based on the technology selection, technology development.

We will continue coordination with DOE to resolve key technical issues and also underlying policy issues. Also to close the R&D gaps.

And we plan to come before you hopefully this time in a more frequent interval than had been in the past. We'll just wait for your, you know, design indication and we'll come back.

CHAIRMAN BLEY: Thank you, Sud.

On the digital instrumentation and the human factors, I just want to mention I know you folks all talk to each other, I'm sure, all the time. But later this week, the human factors folks from Research are going to present to us work that was done with some focus on the possible negative impacts of some of the new instrumentation on human performance in the plants.

I hope that's getting factored into the work you folks are doing as well.

MR. BASU: Let me see if I can get Steve's attention.

Steve, would you address that particular item in your presentation this afternoon?

MR. FLEGER: Yes, that work you're referring to, I believe, is the degraded I&C.

CHAIRMAN BLEY: Yes.

MR. FLEGER: The interface issues. I'll just briefly mention that this afternoon.

CHAIRMAN BLEY: Okay.

MR. FLEGER: But this Friday there will be a lot more detail on that.

CHAIRMAN BLEY: Okay. I just wanted to make sure that that program was feeding your program.

MR. FLEGER: Yes, sir. Yes, it is.

CHAIRMAN BLEY: Okay. Thank you.

MR. BASU: Thank you. I think that's all I have for the overview.

Any question, comment?

CHAIRMAN BLEY: Anything from the Committee before we get to the details?

MEMBER RAY: Well, again, having served on a HTGR Codes and Standards Committee in the past, that was one of the things listed up here, I just at some time would like - I mean, we take for granted that nobody is going to redo things that were done before. And that's an assumption that I think doesn't need to be tested, even.

But it just seems like there is a starting point there. So, you're not starting from complete scratch as you carry this forward in that there was a plant built and operated. Moisture intrusion, for example, was a big deal. And I just would like some mention of how the work that you're talking about doing now builds on what was done at that time and how the licensing basis, for example, at Fort St. Vrain would be relevant.

MR. BASU: Very good point. We are going to make most use of the legacy information and data in drafting the -

MEMBER RAY: I would take that for granted, sure.

MR. BASU: Any other -

CHAIRMAN BLEY: We can go on to the next topic.

MR. BASU: Thank you, again. So, I'm going to hand it over to Joe Kelly who is - there he is.

(Off-record discussion.)

MR. KELLY: Okay. So, my name is Joe Kelly, and I'll be talking about the NRC development of its evaluation model for the NGNP.

And, actually, I have ninety minutes, but five presentations. So, I will be giving a, hopefully, brief overview of the evaluation model itself.

I'll be followed by Mourad Aissa talking about the SCALE development that's our nuclear analysis tool. Then by Hossein Esmaili talking about MELCOR development for NGNP. And that's in the accident analysis area.

Then I'll come back and talk about core analysis. That's the PARCS/AGREE codes, and then the NRC-supporting experimental programs.

So, the objective of this work is to give us a confirmatory safety analysis capability, basically, an evaluation model, as to both supports the NGNP licensing review, as well as provide a technical basis for regulatory decisions.

So, what's the scope of this? What is it that we have to do? What are we trying to compute?

What we want to do is be able to evaluate the radiological consequences. And that's basically just to dose to the public and the workers.

To do that, we have to be able to calculate the fission product release from the containment and its subsequent dispersal in the atmosphere.

So, consequently, we need expertise in five different areas; nuclear analysis, thermo-fluids, fuel performance, fission product transport and consequence analysis.

And so, all of that's going to be part of the evaluation model that I'm going to talk about today.

As Sud said, up to now it's generic in scope. So, we have to apply it to both the pebble-bed and the prismatic. And those are the abbreviations I'll be using for pebble-bed and prismatic.

And, traditionally, an evaluation model for a gas reactor has three separate evaluation models that we'll be rolling into one.

The first part is normal operations. What your pre-break conditions are. And those are important because we need to know the magnitude and distribution of the fission products within the helium pressure boundary before an event occurs.

The second part is the initial fission product release or the blow-down release. And then finally the delayed fission product release which occurs during a heat-up event.

This could take more than the ninety minutes that I have if I were to explain every box on this schematic.

So, this is a schematic of the evaluation model. And since I don't have time to go through every box, I'm going to start at the end.

So, if you look at the large box in the bottom right labeled "LBE" or Licensing Basis Event, Transient Analysis, this is really where, you know, our accident analysis occurs.

And then specifically the box in the red on the left, a system accident analysis which is done by the MELCOR code, that gives us our source term release.

That's handed off to the MACCS code for

the consequence analysis. And the eventual product is the environmental release and dose. Most of our transient safety analysis go through that path.

On the other side of that box you'll see the PARCS and AGREE codes listed. Those are basically for any type of reactivity insertion event that requires a 3D kinetics code to be able to resolve it.

The rest of this, the other three main boxes, all had to do with normal operation. And that's a little bit of a disconnect from what we do with Light Water Reactor where, you know, primary focus is loss of coolant, you know, whether it's large break, small break, whatever.

Here, a lot of the focus is normal operation. And as I said, that's because we need to know both the magnitude and distribution of the fission products within the system over the life of the plant.

So, to do that, the first thing we need to do is, you know, be able to calculate the steady state of the plant. And to do that, we need the box labeled "Nuclear Data Preprocessing."

This is the province of the SCALE code. Mourad is going to talk about that right after me. So, I'm not going to say much there.

But one of the primary outputs of that - let's see. I don't have a pointer here. So, few-group cross-sections.

And in this case for, you know, modern-day calculations for gas reactors, we're talking about like 23 to 26 groups. So, it's not the two to four groups that you might be more familiar with, with Light Water Reactors.

So, those are handed off to the box labeled "Normal Operation." And PARCS is our core neutronics simulation code. AGREE is a thermo-fluids module for gas reactors.

I have a presentation on those. So, I won't say too much in this one about those, but they're intimately coupled.

CHAIRMAN BLEY: Are these codes all appropriate now for doing these calculations or are you having to modify most of them?

MR. KELLY: Well, all of the codes with the exception of AGREE are the NRC Light Water Reactor codes.

CHAIRMAN BLEY: Right.

MR. KELLY: So, we're modifying all of them. And I'll discuss what some of the main modifications are later in my presentation.

CHAIRMAN BLEY: Okay. And if you could at some point, mention where you stand in that and what the schedule looks like for it.

MR. KELLY: Okay. At the top of the Normal Operation box, there's a little box that says "Driver (SNAP Plugin)."

And the reason we have a driver here is because we need to not just have one steady state, but have the evolution of that steady state over the fuel cycle whether you're searching for an equilibrium core in a pebble-bed or whether you're following the operation of a prismatic as you do the fuel shuffling and the cooling of the control rods. So, that box is a little more complicated than normal.

One of the main outputs from that is the power distributions. And those along with bypass flow go over to the box called "Fission Product Preprocessing."

And so what the job of this is, is to determine the magnitude and distribution of those fission products by first calculating the core-wide fission product release - and that will be done with MELCOR - and then the distribution of those released fission products within the system.

And Hossein is going to talk about that.

So, I'll pretty much skip over it.

MEMBER CORRADINI: Can I go back to your box -

MR. KELLY: Yes.

MEMBER CORRADINI: -- just so I understand?

So, the thing that always concerns me about the gas reactor is the change in geometry with irradiation.

So, where is that being done or is that left to DOE?

MR. KELLY: Well, that's a good question, actually. One I glossed over.

There's a box labeled "Thermal & Irradiation Geometry Changes." And of course the main reason that's important is because of its affect on bypass flow.

Bypass flow then determines your peak fuel temperatures. The peak fuel temperatures affect the fission product release during normal operation. So, that's something we have to handle.

To do that calculation, you need three parts. One part is a numerical scheme to handle bypass flow. I'll show that in the talk on PARCS/AGREE.

The second part are the constitutive

models for the loss coefficients, et cetera. And I'll show you something on the supporting experimental programs for that.

But as Mike said, the more important part is actually how much does the prismatic - well, let me just say "prismatic" for the moment - distort?

Well, to get those bypass gaps, typically there are three different things. The first is the as-built gaps. Which in the core, are on the order of one millimeter.

Then there is the differential thermal expansion. They're designed to try to bathe all the metallic components with helium at T-inlet. So, it's like 260 degrees C.

But even with that, the metallic components expand a lot more than the graphite components and are, in fact, pulling apart the fuel element stacks a little bit and opening those gaps up to something in the order of two millimeters.

Then you have the irradiation-induced damage, and this is the big one. Now, later on today Srini is going to talk about graphite. And he does have a program at Argonne National Labs to do a irradiation damage structural analysis of one fuel element.

Actually, trying to model all of the fuel elements in the core together, that's a huge research program that we at the NRC are not going to be able to do. DOE or Idaho is trying to develop a full-core model, but that's in the future.

The more traditional method is to model each assembly by itself and calculate the distortions of that.

Srini will be giving us, if you will, a numerical benchmark. So, a number of different calculations at different, you know, thermal gradients, different fluences and fluence gradients.

But then as part of the evaluation model, we'll have to have a simpler way of calculating that. So, it would be something like a multi-grain problem.

MEMBER CORRADINI: But to make sure I understand, so if I knew the gradient and if I knew the irradiation history, I could with the proper graphite, predict the swell or the change in dimension.

Is that a given?

MR. KELLY: Well, once we have the graphites and once the AGC program is done, yes.

MEMBER CORRADINI: Okay. So, then it's not so much that as -- so, you have the benchmark there.

It's a matter of putting it together with multiple blocks or multiple things under a known temperature gradient and irradiation history.

MR. KELLY: Right. And then you have these blocks stacked on top of each other, each one distorted a little bit.

And as you know at the reflector boundary where the fluence gradient is very sharp, one side of a block can shrink more than the other side. So, then you can open up web-shaped gaps and the possibility for the columns to bow.

But actually trying to calculate the interference of one column with the next is a huge research program, and that's something that Idaho is working on. It's beyond the scope of something that we're able to do here.

So, then you have to say, well, how can we, if you will, simulate all of these little wedge-shaped gaps?

Do we close them all up and have big ones top and bottom? Do we distribute them? And that's probably going to be done in some kind of probabilistic way, but we're not there yet.

MEMBER CORRADINI: Okay.

MEMBER ARMIJO: Well, these issues existed for Fort St. Vrain.

MR. KELLY: Yes.

MEMBER ARMIJO: And they were addressed somehow in the regulatory process. And at some point going back to Harold's point, I'd like to know what we learned from that and what the deficiencies were at that time that we're resolving with this new research.

MEMBER CORRADINI: Or they just took a very -- equivalent of a very bad hot channel factor and essentially have to live with a large margin.

MEMBER ARMIJO: Sure.

MEMBER CORRADINI: Right.

MR. KELLY: In the licensing basis, I don't know. And that's something that NRO will be looking at.

I did learn how GA went about calculating those gaps, which is why I know the little bit that I do know on that subject.

Because as Mike noticed, I know more about two-phased flow than I know about gas reactors. I'm learning.

CONSULTANT KRESS: Before you leave, one other question on that box.

The pebble-bed reactor has spheres flowing by gravity down through it going in at the top and coming out at the bottom.

That's a complicated flow pattern and I'm worried about spheres getting stuck in there and not moving and things like that.

I didn't see anything in the plan that talked about actual experiments, a picture with real pebbles that may address the modeling for that.

Do we have models for that kind of flow through?

MR. KELLY: Well, people are using discrete element mechanics codes to model pebble flow. And that's - the Idaho Lab has their own version called - I think it's called PEBBLES.

CONSULTANT KRESS: Is this a probabilistic model?

MR. KELLY: No, it actually calculates all the forces between each and every pebble and the rolling friction and so on. But of course you need the sliding friction factors of the graphite at pressure and temperature, and that has to come -

CONSULTANT KRESS: And they may change in dimensions also.

MR. KELLY: Yes, they'll probably change more in dimensions due to, you know, the -

CONSULTANT KRESS: But are there plans for experiments to look at this or are we just going to -

MR. KELLY: Not at the NRC. That would have to come from the applicant or DOE.

MEMBER REMPE: Joe, could you kind of just clarify this is like a joint DOE-NRC analysis approach or is this the NRC one? And then what does industry do, and just kind of give us an overview of the different ways people have been —

MR. KELLY: This is the NRC one.

MEMBER REMPE: Entirely.

MR. KELLY: Entirely.

MEMBER REMPE: Not using any of the DOE analysis.

MR. KELLY: No.

MEMBER CORRADINI: So, just to push that point since some of us will be asking that of the other side, so the blue is NRC.

So, there's a red version of this that DOE has that will be handed off to the industry, or you don't really care about that. Somebody is going to provide you a red version of this.

MR. KELLY: Yes.

MEMBER CORRADINI: Okay.

MR. KELLY: And we'll be using this to, you know, check those - to audit those -

MEMBER REMPE: There will be some back and forth.

MEMBER CORRADINI: Right.

MEMBER REMPE: And how that's going to work is DOE's business, and I don't know.

MEMBER CORRADINI: Okay.

CHAIRMAN BLEY: Let me follow up Tom's question.

MR. KELLY: Sure.

CHAIRMAN BLEY: If applicant is happy with detailed calculations for this possibility of the pellets/balls not moving like you'd expect them to, do you foresee NRC needing to see some kind of experimental verification or does that kind of modeling strike you as complete enough to be confident that we won't have significant problems in that area?

MR. KELLY: Well, see, it depends. It depends on a lot of things.

If you look at the history of something like THTR or ABR, they had very different pebble flow profiles inferred from their burn-up than they actually, you know, thought they were going to have ahead of time.

And from the THTR, they did have problems or appeared to have problems with pebbles getting hung up, but part of that's because they were crushing pebbles by inserting the fuel rods down through the core, you know, basically a hydraulic ram on the control rod. So, that's not a good idea to have large pebble fragments in your discharge chutes.

How important it is, we don't yet know. Our calculational tools are not yet at the point where we can do a lot of sensitivity analysis to check it.

That's one of the things we will be doing. And depending upon how important it ends up being, then the requirements for what an applicant might have to show us will change.

Now, for example, the initial - not the initial, but when I started this, the pebble-bed design was the PBMR-400 and that's an annular core. Relatively thin annular core.

So, if you talk about a radial pebble velocity profile, well - but you had to worry about an azimuthal one. And I don't know of any data on an azimuthal velocity profile, but that design has since, if you will, devolved back to something that looks a lot more like the old HTR model, which is a cylindrical core. And so, now you're back to a radial flow which we have more experience with.

And Sud alluded to the possibility of an agreement between us and China on exchanging data, you know, tools for data with HTR-10.

And so, if you can get burn-up data on the pebbles coming out of HTR-10, then we've got a pretty good idea of what's actually happening in a real reactor under prototypic conditions.

So, you know, we're not - I don't have the answers yet.

MEMBER ARMIJO: Does NRC have the data from the ABR on their burn-ups and, you know, they operated for a long time. And so, they should have had a lot of data.

MR. KELLY: Yes, I think we do have some of it. But as you know, the AVR has a very long history, but it's also a very complicated history. Many, many different fuel types and things like water ingress events.

But you're right. That's one of the starting points to look at, yes.

MEMBER ARMIJO: Okay.

CONSULTANT KRESS: The burn-up data, can that be done - is that going to be done online?

MR. KELLY: Yes. They have a burn-up measurement system. So, when the pebbles roll out -

CONSULTANT KRESS: Specially at cesium, for example, that would tell you how long -

MR. KELLY: Dave Petti is shaking his head. So, I'll go with Dave.

MR. PETTI: Tom, it's a cesium to a nonvolatile like psyllium, something like that. They have that ration and they've got a very tight window, you know, to make the measurement and make the decision.

CONSULTANT KRESS: That would tell you how long the pebbles have been in -

MR. PETTI: Right, right.

MR. KELLY: So, when you break that down, these are the tasks we have. The first is code and model development. And that's where we've been spending most of our time today. And that's pretty well along, and hopefully I'll be able to give you an idea of some of that in the course of these five presentations.

The next is code integration. And that's to put all those codes together into an automated workflow.

Third is an uncertainty analysis, because we plan to do this as best estimate plus uncertainty. And we'll be using a statistical approach. We're just starting a task on that now.

Finally, code assessment. And we're likewise just beginning that now to assess the MELCOR code against the existing database. And that will continue between now and the end of the program. And the final product is of course a validated EM with a code applicability report.

So, MELCOR is the NRC severe accident code. It gives us a two-dimensional flow, heat transfer and fission product transport capability.

I'm not going to say too much about this because Hossein is going to come along in just a few minutes, but you can see the completed development tasks and then the ongoing development tasks on this slide and I'll just make a couple of points.

The first is one that's completed is new core models. One for the pebble-bed and one for the prismatic. And, actually, I'm going to show a very quick example for the prismatic.

They've also installed graphite oxidation models and what's called a stratified counter-current flow model for air ingress.

I won't be showing the MELCOR model for that, but I will be showing you a slide from an experiment on that.

CONSULTANT KRESS: The graphite oxidation, is that for water ingress and air ingress?

MR. KELLY: This is for air ingress. The water ingress is something that's going to be addressed.

CONSULTANT KRESS: Because the water will react with the graphite also.

MR. KELLY: Yes.

And then for the ongoing tasks, and this is what Hossein will be talking about where most of the effort is going today, is the fission product release and transport. And then future tasks are things like extending the aerosol model to be able to model dust.

So, this is a very simple - it's the result of a numerical benchmark. So, this is the new - or PMR core model.

And so, what we're looking at is a conduction problem at what we call the Meso-Scale. So, we're talking about the heat transfer resistance between the fuel compact and the coolant channel. So, going across the graphite web to the coolant channel.

This is important for normal operation. To get this right is important for normal operation, because it sets your fuel temperatures and your moderator temperature.

Well, I said MELCOR is a 2D code, and you can see the unit cell it uses for this. This is its core model. So, it's basically a fuel rod with a gas gap.

The clad in this case is now the graphite moderator. So, that's what's going to be calculated in MELCOR.

On the right, I have a finite element model of a unit cell taken out of a prismatic block. So, in each prismatic block there are two fuel compacts for every coolant channel.

And they come in this if you make a little right triangle here with a sixty-degree angle and a thirty-degree model, but I modeled the whole one here.

So, you have the fuel compact, a small gas gap, and then the three coolant channels. And for the purposes of a numerical benchmark, we're specifying the gas temperature and heat transfer coefficient to be the same between MELCOR and the finite element calculation.

And so, what we're trying to do is see how well this relatively course 1D conduction model - 1D/one-parameter conduction model actually does a 2D problem.

And so what we get from MELCOR is the

average fuel temperature and the average moderator temperature. And you can see the discrepancies.

This is with a radial power factor of one. It's only a couple of degrees. It's surprisingly good. So, that's good news.

The SCALE/AMPX, this is our nuclear analysis code suite. And this is what Mourad, who will be the next speaker, will be talking about.

So, this is where we go from the in-depth nuclear data libraries to actual cross-sections that can be used in the PARCS code.

And so, SCALE gives us the lattice physics capability, as well as depletion capabilities. So, that gives us what we're calling few-group, or I should say multi-group cross-sections. The decay heat, as well as the fission product inventory.

One of the big things - accomplishments in that was being able to model the double-het model for both the pebble-bed and the prismatic fuel, and Mourad can talk about that.

A lot of the current work is on doing benchmarking both the HTTR and HTR-10 primarily, and Mourad will show some of that.

And as well as improving and validating the interface to the PARCS code, and I'll show some of that when I get to the PARCS/AGREE presentation.

I said I'm going to have a presentation on PARCS-AGREE. Since I'm already running overtime, I'll skip this slide and talk about them when I get to that.

SNAP, which stands for Symbolic Nuclear Analysis Program, is the graphical user interface for the NRC codes. And so, we will be using it in the normal, traditional way for both pre and post-processing for MELCOR and PARCS/AGREE.

But we're going to be using it a lot of other ways as well, because it has an extensive plugin capability. So, as I mentioned, we'll be using it as the driver code for things like a search for the equilibrium core in a pebble-bed, or for handling the fuel shuffling methodology in a prismatic.

We'll also be using it as the glue to stick all these codes together. So, when you have to take the output from one code and massage it to make it be the input for the next code in a chain of calculations, all that will be done by a SNAP plugin instead of human intervention.

It also has what's known as an auto validation tool. So, when we do code assessment, we don't expect to just do it once because, you know, you will come up with code deficiencies, you'll remediate them and have to repeat the calculation.

This will allow us to repeat those calculations very easily to minimize the engineering time that is done, as well as due to the uncertainty analysis tool.

And so, the code and model development tasks are well underway. As we go through each of the next four - three presentations, you'll get a better idea of that.

Preliminary code assessment against the existing database is beginning now, and we plan to have our independent confirmatory analysis capability ready by the end of 2013.

And so with that, I'll hand off to Mourad.

CHAIRMAN BLEY: Okay. Thank you.

MR. KELLY: You're welcome.

CHAIRMAN BLEY: You're right. That did take a little longer. Are some of these - are they all five about the same length or is it some of them will be shorter?

MR. KELLY: Well -

CHAIRMAN BLEY: We need to pick it up just a little bit, or we're going to stretch on a little late.

MR. KELLY: Okay.

(Off-record discussion.)

MR. AISSA: Good morning. My name is Mourad Aissa. I will be talking about the neutronics analysis part of our campaign.

First, the big picture, and Joe already talked a little bit about this process. We'll start with the ENDF data. Right now it's the ENDF-7.

And the AMPX 2000 generates your problem independent cross-section, but are continuous for the stochastic codes and the multi-group for the deterministic codes.

Right now we have one that's 238 groups at which we devote most of the validation results that I will be presenting. And we have an alternative 81 groups that we are most getting ready for production mode.

TRITON is one of the control modules in SCALE. SCALE has several control modules. So, one can insert SCALE for TRITON for the purpose of this talk, the equivalent.

Those planned cross-sections are processed for resonance by BONAMI and CENTRM for any resolve resonances and resolve organizes respectively. BONAMI does the end result. CENTRM does data result.

Once you prep your cross section then you go to the lattice codes. One is 1D for PPR XSDRN, and NEWT is 2D for the prismatic. And you get your sets of few-group cross-section that you can use by the nodal core simulator. In this case, PARCS.

Some of the major activities that we've been involved in, like I say, we just finalized an 81 energy group which is optimized for HTGR spectra. We already have one working group for 20, 30 years because it takes longer to process problems with a finer energy cross-section.

The multi-group resonance processing methods have been improved successively. We had the double-heterogeneity (DH) self-shielding method that was based on running CENTRM, which is the result resonance core. And it was excessively taxing time-wise.

There was an alternative method that's based on intermediate resonance, basically assuming narrow resonance treatment of the - below unresolved resonance region. The narrow resonance would give you better results and also speed up the execution by a factor of four to seven.

The other item of note is we included a continuous energy, KENO, which is a Monte Carlo

Stochastic code. And in it, we included the resonance upscattering treatment for the U-238, which improves the k-effective by 200 to 400 pcm. And also, this would be added to CENTRM.

The lattice physics code, before we had only NEWT that does lattice in a rectangular geometry.

We kept NEWT and we made it functional for the hexagonal geometry of the prismatic and improved quadrature sets to allow hexagonal reflections, and accelerated the CMDF, which of course masked the difference to address the slow convergence of the graphite reactors.

And for the PBR, we added to the TRITON the XSDRNPM, which is also a discrete ordinate transport code just like NEWT.

The Monte Carlo depletion along with the continuous energy, KENO, we have a multi-group KENO which will test those cross-sections that we develop for use for PARCS later on.

There is the benchmark. First, this is a fuel block for HTGR. And there's more of a code-to-code validation or comparison.

The KENO continuous energy and the multi-group are compared to MCNP5 continuous energy. As you see, there really are acceptable results. About 85 pcm between the two continuous energy codes. And 336 pcm for the multi-group, which is an acceptable result.

MEMBER CORRADINI: I don't know what I'm looking at. I think I do, but I'm really not sure.

So, that's a cartoon prismatic block on the left?

MR. AISSA: It's just a block, yes.

MEMBER CORRADINI: What's the thing on the right?

MR. AISSA: This is a KENO model of it.

MEMBER CORRADINI: Of what, though? I guess I don't know of what. The thing on the right is a model of what?

MR. AISSA: No, no, I'm sorry. This is just the schematic diagram of the annular fuel pellet. So, this is just like it says. The TRISO that has the particles, and the helium channels that - it's basically just a blowup of the -

MEMBER CORRADINI: But I guess -- I'm sorry. This is not relevant to our discussion. I was just trying to understand.

So, the prismatic block doesn't look like that. So, is there a homogenization, is there a modeling simplification that's made to do this?

Because you've got the - you've got the fuel compacts, you've got the helium, you've got the graphite between them.

You've got to do some sort of heterogeneous calculation, homogenize that.

MR. AISSA: Yes.

MEMBER CORRADINI: And then take that homogenized thing and grow it to a bigger homogenized block.

MR. AISSA: Yes.

MEMBER CORRADINI: So, is this a benchmark or -

MR. AISSA: No, no, is not a benchmark. Is just a geometric representation of the actual physical condition of it because it's just -- it shows the -

MR. PETTI: Let me just point out to the panel, not to be confused, HTTR has a prismatic configuration that is not like NGNP or any of the US prismatic. It's an annular fuel.

Okay. So, --

MEMBER CORRADINI: Those are unique.

MR. PETTI: Right. It's unique, but it's what we have the benchmark on. So, it's the right thing to do.

MEMBER CORRADINI: Thank you. Thank you.

MR. PETTI: So, don't get confused on the geometry.

MR. AISSA: The fuel pellet has a hole in it. I'm sorry about that. I didn't realize.

MEMBER CORRADINI: Thank you, Dr. Petti.

MR. AISSA: And for the full result, this is actual experiment based on the IAEA benchmark.

And in here, we don't have as good results because as we see, the - both MCNP and KENO have a bias between 1.5 to 2.0 percent k-effective. And this has been also observed by other people who did the benchmark.

It's still being analyzed, but we suspect maybe that the graphite has some impurities, boring impurities. And other people are saying maybe the ENDF carbon cross-sections in the thermal region is not captured right.

And some people say when you use the JNDL, which is the Japanese Nuclear Data Library, you should get a little better results, but this we still are working on it. But I don't think it will move too -

CHAIRMAN BLEY: Before you go, let me ask a naive question.

So, these are one experiment and three calculations on k-effective?

MR. AISSA: Yes.

CHAIRMAN BLEY: With sigmas that imply they're tracking the uncertainty, but they don't fall within each others' bounds. So, the sigmas are in some fashion, a great underestimate of the uncertainty, it appears to me, unless I'm too naive in the way I'm looking at this.

MCNP5 has a sigma of 0.0001, and yet it's off by a factor by 0.02 from the experiment. So, the sigmas don't seem related to any real uncertainty. There are some other -

MR. AISSA: Actually, the way it was given to, actually it's off by more than that. It's off by 2200 pcm.

CHAIRMAN BLEY: Right.

MR. AISSA: Yes.

CHAIRMAN BLEY: So, what are the sigmas? What kind of uncertainty are they talking about? is that just an uncertainty in the calculation?

MR. AISSA: The sigma, that's your code gives you.

CHAIRMAN BLEY: The code gives you that?

MEMBER CORRADINI: At least -

(Simultaneous speaking.)

MR. AISSA: It's a statistical.

MEMBER CORRADINI: Okay.

CHAIRMAN BLEY: Okay.

MR. AISSA: They used -

CHAIRMAN BLEY: So, it's an uncertainty on the calculated number from the trials.

MR. AISSA: Yes, yes.

CHAIRMAN BLEY: I don't know what the sigma on the experiment is. That's not on trials. That's some other sigma.

MR. AISSA: That is in the benchmark, the IAEA benchmark is what that was given.

MEMBER CORRADINI: But does that mean that that's the sigma of actually a set of multiple measurements of k-effective?

What does that mean? I think that's what Dennis is asking.

CHAIRMAN BLEY: Yes, what does that sigma stand for?

I get the one that's on the calculations now.

MR. AISSA: It is a measurement of uncertainty. How many measurements does it capture? I don't know.

MR. PETTI: It's the Japanese team's estimate of the uncertainties of the experiments that were done. All the classical experimental uncertainties aggregated.

CHAIRMAN BLEY: Okay. Thank you.

MR. AISSA: Now, for further results, the control rod worth results were -- show a very good agreement between the experiment and the KENO CE and KENO MG.

The temperature coefficient results that are plotted here, many can see the - probably that will answer some of your questions about uncertainty measurement.

As you see at 150 C, there was a huge measurement uncertainty that - this would show very good agreement between MCNP and SCALE whereas the measurements are just had a little bit.

The shutdown margin results for all control rod in, there was very good agreement. When the reflector control rods are in, there was a bias where we have less by three percent k-effective.

Moving on to HTR-10 first critical core benchmark results, those - basically, the measured was as close to one.

And this would -- looking at what the MCNP or the International Reactor Physics Evaluation Group has come up with and compared it to our SCALE/KENO and with very good agreement. 121 pcm. And just comparing our KENO to our MCNP, that's minus 162 pcm. So, those are also acceptable results.

The PROTEUS benchmark results, we are still working on them. Those are experiments taken at the Paul Scherrer Institute in Switzerland. and PROTEUS is a very flexible experiment and it was reconfigured for pebble-bed reactor.

And I see 1, 1A, 1A(2) and 2, 3. Those are configurations of how many combinations of fuel pebble, modulator pebble.

And as you see, those also show acceptable results where the average delta k-effective is 252 pcm.

SCALE-PARCS integration, Joe is going to talk a little bit more about the detail of some of the PARCS pictures.

And, also, Joe mentioned that SCALE provides sets of cross-sections that are in 23 groups, which is compared to four groups to LWR or even two groups for LWR cases.

And, also, we're doing some work for HTR-10 to ensure --just a sanity check. There were exercises -- sets of exercises using block models and 2D core models that were run both by SCALE and by

PARCS with acceptable or very good agreement.

Ongoing activities, we complete PROTEUS validation work. And, also, we're still working on the HTGR benchmark, the one -- I'll show you.

Publish NUREG/CR on HTGR validation with SCALE addressing the three experimental sources. Complete SCALE-PARCS integration.

Continue work on space-dependent reaction rates in both the - engage in any kind of reflectors, evaluate code prediction biases and uncertainties, and update Version 6.1 of SCALE code system and documentation to be released for fission. Thank you.

CHAIRMAN BLEY: Okay. I just want to make one comment to you.

That one chart really bothered me. It looks like a presentation that's showing uncertainties in calculations when it's really not. Those things mean different things.

MR. AISSA: Which one?

CHAIRMAN BLEY: the one I asked you about. Slide 6, but you don't need to go to it.

MR. AISSA: Yes.

CHAIRMAN BLEY: It's just the sigmas mean very different things and they're talking about different things.

The ones on the code aren't estimates of the uncertainty in the calculation. They're estimates in how well you're coalescing to the value in the - they're very different things.

MR. AISSA: Yes.

CHAIRMAN BLEY: And it gives the impression that they're really uncertainty in the result like the uncertainty in the measurement.

So, I think you need to think about how you present uncertainties in the future.

MR. AISSA: I agree. That could be misleading and the point is taken.

CHAIRMAN BLEY: Thank you. I think we better move on unless somebody wants to - we're catching up a little bit. We're five minutes behind, I think.

Thank you.

MR. AISSA: Thank you.

CHAIRMAN BLEY: Do you have your next - you can find your name tag while he's setting you up.

(Off-record discussion.)

MR. ESMAILI: Okay. As Joe said, my name is Hossein Esmaili. I'm in the Office of Research. As Joe said, I'm going to be mainly focusing on MELCOR fission product modeling, what we are doing right now and provide you with some status report on where we are at.

The main objective of the present work that we are doing is to develop HTGR-specific models for fission product release and transport.

We already have MELCOR. It's a system level code. We already have the infrastructure. So, the idea was to use the existing MELCOR models for transport and deposition in primary system and into containment, but develop additional models as needed to answer specific HTGR phenomena.

These are develop diffusional release models for TRISO fuel particles, develop diffusional release models for release of fission products through the matrix and graphite block for the prismatic, and also develop additional transport model. Mainly, lift-off, you know, this is important for HTGR. We don't have to live with that in LWR. Turbulent deposition is important in pipes and agglomeration, I'm going to get into that, and most importantly dust, you know, dust generation and deposition.

The models that we are developing has to be applicable to both pebble-bed and prismatic designs. And the code should be able to calculate releases both during normal operation and during accident conditions.

MEMBER ARMIJO: All right. In the release models, you say diffusional release.

Do you have any other sub-model to address the releases for, let's say, fractured or imperfect TRISO fuel particles?

Does this analysis assume that every TRISO particle is sound as designed or -

MR. ESMAILI: No, we do have - well, the - I can get into that a little bit later, but we do have models for intact particles, you know, all the layers are there, models where only it's a bare kernel, none of the layers are there, or we can do any combination of -

MEMBER ARMIJO: Okay. So, you have several models and you can -

MR. ESMAILI: That's right.

MEMBER ARMIJO: -- mix as you -

MR. ESMAILI: That's right. We have that capability.

Okay. So, this is an example of what not to put on a viewgraph, but I just wanted to make a point here that in MELCOR space we do have, you know, different packages.

What I'm referring to here is that how we are doing things in what is called a radionuclide package.

So, at the start of the timestep, we calculate releases, you know, fuel from the core. In the case of LWR, from the cavity. And what you see on the left-hand side is what we are changing in the infrastructure of the code.

In this case for LWRs, we are using, you know, CORSOR-Booth model because that's, you know, the time of interest is only of a few hours or more.

But in HTGR we care about, you know, during the normal operation and accident conditions. So, here we are doing a more detailed model of diffusion for all the layers.

After that, you know, you already have aerosol dynamics model. That takes care of, you know, how the fission products gets out of the fuel, you know, they can condense from aerosol deposit.

So, these are all based on MAEROS equations. They have already been integrated into the code. So, we are using most of the, you know, the models that's already in there, but we are developing, as I said before, a turbulent deposition model, a lift-off model, you know, during depressurized loss of coolant accidents, and dust generation.

Dust generation since we don't have much data or a good, you know, first-order model, we are going to rely on a parametric model, you know. Dust would be formed as an aerosol and injected into the air.

CONSULTANT KRESS: So, it's amount and -

MR. ESMAILI: And the amount and -

CONSULTANT KRESS: -- size.

MR. ESMAILI: That's right. That's right.

CHAIRMAN BLEY: An administrative question, not a technical one, but we're calling these modified Light Water Reactor codes by their old names, and they're quite different. I'm just wondering if we're going to get into confusion in the future.

Are you going to rename them as you go along?

MR. ESMAILI: No, because MELCOR is one code. So, all you have to do is that, you know, we are not - we don't have a MELCOR LWR version and a MELCOR HTGR version. It's all in a single code.

The only thing you have to do, if you specify what type of reactor you're modeling. Right now we do have -

CHAIRMAN BLEY: The whole thing is built into the one code?

MR. ESMAILI: That's right. Once the user says, you know, I have a BWR, PWR, PBR or PMR, the code already goes and adjusts and uses the appropriate models.

CHAIRMAN BLEY: Okay.

MR. ESMAILI: So, the whole idea of MELCOR is that it has to be an integrated code so we don't want to -- make sure we know we have QA and, you know, version control and everything.

CONSULTANT KRESS: I mean, are you using the same fission product groups that -

MR. ESMAILI: That's right. The whole idea is that we are using the fission product classes -

CONSULTANT KRESS: Same species.

MR. ESMAILI: Same species. But as you know, you know, we can generate our own classes. So, for certain amount of, you know, isotopes like cesium-137, we are going to track them individual. So, we built different classes, okay, for certain isotopes. The rest of them we are grouping together.

But for example, you know, cesium-137 would have it's own class with everything. All the other isotopes, you know, would be in the class of, you know, cesium class.

So, you know, we have flexibility how we can do this class transfers or, you know.

MEMBER CORRADINI: So, let me ask a different question. So, you can hit a button and the code knows what to do, in theory.

Do you freeze it so that you can't accidentally do something wrong?

CHAIRMAN BLEY: Or does the output clearly tell you which modules this went through so you know you got the right calculation?

MEMBER CORRADINI: If you think back where I'm going from, Hossein, if you think back when they were doing the IPEs, the map basically started as a map for a B and a map for a P, and then eventually it came together.

But in some sense, the fact that they separated allowed you not to potentially muck it up, so to speak.

Is there a QA control here such as you don't accidentally cross boundaries?

MR. ESMAILI: Absolutely.

MEMBER CORRADINI: Okay.

MR. ESMAILI: Everything is under, you know, QA control. And as a matter of fact, the people who are working on the LWR version, they're, you know, so, they can see whoever is working on the code. So, you know, you're not going to have that problem.

Again, you know, we have, you know, we have done this in the past, you know. We have MELCOR PWR and BWR which was already done with one keyword in the input.

Okay. So, water and fission product condensation, evaporation, you know, these are already done in the code. We don't really necessarily have to change anything.

Auxiliary models, filters, spray, you know, you don't have to worry about iodine pool, but we don't have to - additional models is not needed.

And aerosol and vapor transport within control volumes, this is already done in the code. So, we don't need any specific models for HTGR.

MEMBER CORRADINI: Let me ask another question.

To go back to what Sud said about the designs now are coming back to essentially a steam generator and we have some sort of in-leakage of steam, is there a qualitative difference of the chemistry in the fission product released in the presence of steam, in the absence of steam?

If I had a dry environment versus a wet, a potentially steam environment, does the code recognize this from a chemistry standpoint so that you would just add steam and you'd see a completely different set of things, or is it subtle or is it not different at all?

I don't know enough about the fission products. Do you know what I'm asking?

MR. ESMAILI: Yes. Let me see if I can answer you this way: What we have to do for a dry condition which is an HTGR application, which we didn't have to do for LWR, is that in the LWR space, you know, we only cared about, you know, in terms of aerosol particles, the shaped factor of units. All the particles were spherical.

Right now as I sit here and talk here for HTGR, we could have a range of shape factors. So, this is what we are really changing.

But as far as steam and non-condensibles and other materials, you know, moving through different control volumes or moving through the core, we are using the same modeling approach that we have used -

MEMBER CORRADINI: So, it comes down to the shape factor, but the chemistry I would -

MR. ESMAILI: Chemistry we have - I can get to in a little bit.

MEMBER CORRADINI: Okay. That's fine. That's fine. I just wanted to make sure because the one thing in terms of the design changes that I was wondering would affect this would be the chance that you would have to then again be concerned with some sort of steam ingress into the -

MR. ESMAILI: Air ingress and steam ingress. This is what we are working actually right now.

MEMBER CORRADINI: Okay.

MR. ESMAILI: Okay. So, I don't know whether Joe talked about it, but we do have a model for air ingress right now. And this is basically, you know, we're taking the model that we have developed for LWR for hot-leg natural circulation and we have actually modeled it into the core.

So, we are going to use that to model, you know, air ingress studies. And - okay. As far as input and out requirements, you know, MELCOR is a system level code. It will require some input.

The first is fission product inventory. That's coming from the ORIGEN. So, we have to know what, you know, how much fission products we have.

Fission product diffusion coefficients I'm going to get to a little bit later. So, we have to know what the diffusion coefficients are for each layer.

And core power shape, of course, you know, this has always been an input into the core, you know, because the code is a 2D code. So, we need the radial and axial profile to see how we are going to assign the powers.

And the code is, again, this is a system level code. It's not a fuel performance code. So, we have to rely on empirical data to find out, you know, what the fuel failure rate is. So, this an input into the code.

Dust generation, lift-off and fission products, you know, we have to, you know, we have to put certain models in the code, but eventually we have to compare experiments.

And fission product release under air and water ingress, we are currently working on them right now. But the infrastructure is there. We just have to tweak things.

And as far as fission product speciation and interaction with surfaces, we already have model for chem adsorption. So, we have models for chemical interaction between fission products and surfaces.

We just have to, like, for example, cesium with, you know, stainless steel, et cetera, we just have to make additional models for the HTGR application.

CONSULTANT KRESS: Normally those models talk about a surface that's just a boundary to the passageway. But when you go through graphite, you've got an entirely different surface-to-volume ratio.

I wonder how you deal with - is that dealt with some way when you -

MR. ESMAILI: At this point, what I'm talking about here is that, you know, fission product and dust that gets deposited on the pipe walls and -

CONSULTANT KRESS: Oh, after it gets out of the -

MR. ESMAILI: Yes, once it gets out of the - how it interacts. You know, once it gets to -

CONSULTANT KRESS: So, you're treating it going through graphite as a diffusion process.

MR. ESMAILI: Right, right.

CONSULTANT KRESS: Which may or may not be right.

MR. ESMAILI: That's what we have to - yes.

Okay. The whole idea for this MELCOR fission produce is based on a diffusion model. This is basically a 1D diffusion model. We are applying it both to cylindrical geometries as in case of the prismatic, or spherical geometries as in case of the pebble-bed.

We also use the same equation to model diffusion through the different layers.

CONSULTANT KRESS: Let me ask you a question about that.

Now, the Booth model that's already in MELCOR has an effective diffusion coefficient for cesium.

MR. ESMAILI: Right.

CONSULTANT KRESS: And it calculates the cesium release by that Booth model. And then to get all the other groups, it uses a scaling factor.

MR. ESMAILI: It uses -

CONSULTANT KRESS: It doesn't have a diffusion coefficient for each of the groups.

MR. ESMAILI: Correct.

CONSULTANT KRESS: And my question here is, do you intend to develop a new scaling factor or will you have diffusion coefficients for each fission product group?

MR. ESMAILI: We are - as part of the experiments, we are hoping to get diffusion coefficients for the major isotopes like strontium, cesium.

We have to make a decision once we are going to apply to different isotopes in different classes, then a decision has to be made there, you know.

CONSULTANT KRESS: Yes, I hesitate to think you will get effective Booth coefficients for all of the groups. I don't -

MR. ESMAILI: Well, maybe, you know, they are planning to - they are planning to measure diffusion coefficients for each layer for certain isotopes.

CONSULTANT KRESS: Yes, you could have a different diffusion coefficient for each of your layers.

MR. ESMAILI: That's right.

CONSULTANT KRESS: Or if it was like the Booth models for LWRs, it's an overall effective - it's an empirical coefficient.

MR. ESMAILI: It's an empirical, that's right.

CONSULTANT KRESS: Yes, I wasn't sure whether you were going to give one for each layer, or an overall one.

MR. ESMAILI: No, we are going to get one for each layer. I'll show it to you on the next slide.

MR. PETTI: Tom, the plan that we have gets you the kernel, gets you diffusion coefficients in the matrix and graphite and, in some cases, in some of the layers.

CONSULTANT KRESS: The kernel treats it as one -

MR. PETTI: A d-prime.

CONSULTANT KRESS: Yes, just like the Booth model.

MR. PETTI: Just like a Booth model.

CONSULTANT KRESS: Okay.

MR. PETTI: Yes.

CONSULTANT KRESS: That, I wasn't sure of.

MR PETTI: Yes, that's how -

MEMBER CORRADINI: What are you guys saying over there? I didn't hear that.

MEMBER ARMIJO: I missed that.

MR. PETTI: There's an effective diffusion coefficient for the kernel.

MEMBER CORRADINI: Okay.

MR. PETTI: In the Booth model, there's a diffusion coefficient divided by an A, a nominal length scale squared. And that's sometimes called d-prime, reduce the diffusion coefficient. That's what we will provide -

CONSULTANT KRESS: Would that be the radius in the kernel, you would think?

MR. PETTI: No, I seriously doubt it will be the radius of the kernel because, you know, this stuff goes to really high burn-up fuel. The path length's going to get shorter and shorter over time.

We will be able to track that for fission gases. We'll see that as a function of burn-up very clearly. For the metallics, you just can't - you can't measure it. All you know is how much was released over the entire -

MEMBER CORRADINI: So, your intent just to make sure I understand, the coupling here is that as we've said generally, is you guys are running the experimental data.

But in some sense, in some sense just to get back to it, you're going to do test to failures with various burn-up groups so that you'll know how this diffusion - what did you call it? D-prime? Is going to be affected by burn-up.

MR. PETTI: Yes.

MEMBER CORRADINI: Okay. And then you also then scale off of the elemental radioisotope since you'll have it just for one, or one lead. And then look at the others and scale it off of that.

MR. PETTI: Right. Based on what we understand about the chemistry of the kernel, the chemical form, you know, which groups - the groups are not the same as LWR groups. There will be some slight changes.

For instance, you know, we have to worry about silver, which nobody has to worry about in LWR space.

And so, whether we have truly cesium iodide, for instance, is an interesting question. The data that's out there historically suggests that it behaves like a noble gas.

MEMBER CORRADINI: You mean iodide does.

MR. PETTI: Iodide. So - but, again, in the oxycarbide system, some stuff's carbide, some stuff's oxides. It can change over burn-up.

MEMBER CORRADINI: Okay.

MR. PETTI: We think we understand that well enough to be able to -

MEMBER CORRADINI: But I guess I'm going with this to kind of couple it back to what Hossein is answering.

In some sense, the database that will

essentially nail down the empiricism of this model is the same database that is going to be used and generated by DOE, but also is going to be used by the licensee.

MR. PETTI: Yes.

CONSULTANT KRESS: It's interesting that the Booth modeling MELCOR for LWRs doesn't know anything about burn-up.

MR. PETTI: Right.

CONSULTANT KRESS: You're talking about having Booth models release kernels that include burn-up, because you may have lots of ranges of burn-up.

MR. PETTI: Sure. Yes.

CONSULTANT KRESS: Do you expect to get experimental data for that somehow?

MR. PETTI: The current experiment plan has an irradiation in which we're going to get fuel of a variety of burn-ups with failed fuel. What we call designed to fail. It will fail early so it will leak over its entire irradiation.

CONSULTANT KRESS: Are these in-pile tests?

MR. PETTI: Yes, these are in-pile tests. The AGR tests.

CONSULTANT KRESS: Yes.

MR. PETTI: So, you know, will I have

something at two percent, four percent, eight percent? No. I'll probably have stuff at, you know, ten percent, 12 percent, 14 percent and 16 percent, you know.

Because you've got an axial cosine in the reactor, I can't get the really low - now, the fission gas I'll get continuously. And so, that will tell us a little bit about any changes with burn-up. That will be probably the most interesting, you know.

Whether you decide to modulate based on that for the other ones, you could1 do that, you know. Let's see what the data says.

MEMBER REMPE: While we're talking about data, on your table on Page 4 you talk about a lot of experiments. And in the research plan that was distributed to us earlier, they have a table and they talk about experiments that DOE has proposed and will be doing and then there's - this isn't probably a fair question to you. Maybe it should go back to Sud, but other experiments the NRC will be planning, and will that be discussed today by someone?

MR. ESMAILI: I think Stu is going to speak this afternoon. He has 45 minutes.

MEMBER REMPE: Good.

MR. ESMAILI: I only have twenty minutes.

So, he can give much more detail about the experiments.

MEMBER REMPE: Okay.

MR. ESMAILI: My role is here how we are doing things in MELCOR space.

MEMBER REMPE: But his topic is on fuel performance and fission product behavior, and yet there's a broader range of experiments. And so, I hope that we'll discuss that.

MR. BASU: The experiments I was alluding to in one of my slides, are the experiments that Joe will be talking about in his - one of his presentations.

MEMBER REMPE: Okay.

MR. BASU: And Stu will talk about the INL AGR program a little bit.

MEMBER REMPE: Okay.

MR. ESMAILI: The only other thing I want to say about the Booth model that you mentioned, is that the Booth model is a solution to this diffusion equation anyway. So, I mean, knowing the deal, you know, they both should give the same answer.

CONSULTANT KRESS: I think Dave Petti missed it, but what is your intention for -- there is a use for R in these equations?

MR. ESMAILI: They have to give us the diffusion coefficient and knowing what their, you know, that the reduce diffusion coefficient is D(T) over some A squared. The A square for LWR is some gas bubble in the radius -

CONSULTANT KRESS: Six microns.

MR. ESMAILI: Six microns or what have you.

Once we know that, we can always get the diffusion, you know, diffusion coefficient. That's a constant. I don't think, you know, depending on how they normalize it, you know, we can always get the D back from there, you know.

As we are developing the MELCOR models, we are doing, you know, incremental assessment of, you know, the models that we are putting in for sanity check to see whether, you know, the models are producing the results we want.

So, here I'm showing you some of the results, you know. You look at the third row which is US/SNL, US/Sandia National Lab, this is the MELCOR results.

CONSULTANT KRESS: Are these release fractions?

MR. ESMAILI: These are release fractions. That's correct. These are release fractions. This is particles at a certain temperature.

For example, in Case 1a, it's just a bare kernel. Somebody asked this question before. At 1200 degrees C, you know. In Case 1b, it's just a kernel at 1600 C for 200.

So, these are release fractions and we are comparing code-to-code comparison.

CONSULTANT KRESS: These are release fractions of what? Cesium?

MR. ESMAILI: This is cesium. That's correct. This is cesium.

CONSULTANT KRESS: This is cesium.

MR. ESMAILI: This is cesium-137.

And for this we use, you know, we use the diffusion coefficient that was already published in TECDOC 978, you know. This is based on the old - this is based on the old data. As you mentioned, you're right. This is cesium.

So that you know, INL is a PARFUME code, you know. We try to be as close as possible to the diffusion coefficient. And we are seeing results very, very comparable to the issues that, you know, we can solve the diffusion equation.

Now, I just want to get into, you know, how we are doing this during normal operation. Again, as I said, MELCOR is a system level code.

So, once you create an input that you have heat structure, you have piping, you have, you know, the fission products are going to get deposited, it's very, very difficult to specify how much dust or fission products are going to be on different surfaces in the NQA deck.

So, what we decided to do is an accelerated steady state normal operation type of calculation. So, let the code calculate during normal operation, how much fission products and dust gets released and gets deposited on various surfaces.

That would then be the initial condition for any transient calculation.

CONSULTANT KRESS: What is meant by accelerating?

MR. ESMAILI: By accelerated steady state, we mean that we are not going to go through, you know, like a two-and-a-half year, you're not going to calculate two-and-a-half year or ten year. Ten year, you know, doing burn-up cycle or operating time.

What we are going to do is that we would run the code long enough to find out what the equilibrium conditions are, you know. How much gets released? What is the circulating activity? Because what gets released is getting deposited.

So, we are expecting to get to some equilibrium conditions. And as things are getting released, they're going to get deposited on heat structure. So, you know, that the amount on heat structure increases, which is coming actually on the next - so, if you can do this for certain amount of time, we can just scale it up to, you know, to operating.

So, for ten years, you know, just we are this much more. And this is - this is really up to the user.

So, in the first step, you know, we want to do a system thermal hydraulic. So, we want to get in steady state in terms of core, you know. Core temperature is a function of R and C.

Once we know the core temperatures, then we can run the diffusion calculations, find out what the distribution of fission products are.

So, you can see here on the top in the three-dimensional graph, this is the concentration of cesium. When you saw the diffusion equation in the kernel versus into the buffer layer, do you see that slope? And in the outer layers, it goes almost to zero.

And once we solve this thing for, you know, for burn-up time of about two-and-a-half years, then we know how much has been released.

What gets released goes into the matrix. And for prismatic, it goes to the graphite block and gets finally into the coolant. So, once we solve that, so now we know we have the distribution of the fission product.

The reason we did solve that detailed diffusion equation is to get this distribution to see where everything is. Because during the accident, you know, particles fail. So, you know, whatever is in the buffer now becomes a source into the matrix.

MEMBER ABDEL-KHALIK: Can we go back to the previous slide, please?

Do we understand the reason for the large differences in Case 3a?

MR. ESMAILI: In Case -

MEMBER ABDEL-KHALIK: 3a.

MR. ESMAILI: 3a.

MEMBER ABDEL-KHALIK: Intact, 1600 degrees C, 200 hours.

MR. PETTI: Let me talk to that, Said.

No, I mean, actually we think it's numerical problems in the codes the way they are -

MEMBER ABDEL-KHALIK: Numerical?

MR. PETTI: Numerical problems.

Some of these codes, the German code, for instance is, you know, sort of a 1970s vintage. And so, if you were to show the time dependence, you'd see some differences.

And so, we think it's these, you know, everyone should get the right answer because these are the thought problems. These aren't experiments. These are thought problems.

MEMBER ABDEL-KHALIK: That's right.

MR. PETTI: And they vary, and so we think there's a numerical issue. And so, actually it's something we're going to pick up inside of our Gen-4 collaboration because the IAEA collaboration is complete, but to try to get a better handle on it. So, it's a numerical - these are very difficult problems, because the diffusion coefficients vary, you know, so much lay of the land.

MEMBER ABDEL-KHALIK: But this is the one case that you want everybody to get right.

MR. PETTI: Right. Exactly. Exactly.

MR. ESMAILI: But the important thing is that, you know, when we compare our results with INL making sure that we have all the diffusion coefficients right, we get very, very close results.

MEMBER ABDEL-KHALIK: Well, but -

MR. PETTI: I'm not convinced those calculations are right, you know.

CONSULTANT KRESS: Those are such small -

MR. PETTI: Small numbers, yes.

CONSULTANT KRESS: You wouldn't expect the error there to really overwhelm everything.

MR. ESMAILI: What we have to do is that, you know, in the silicon carbide layer, you know, we play around with the meshing, you know, that that can affect it.

MR. PETTI: It's actually the kernel - we found the kernel meshing is absolutely critical. You need a heck of a lot of nodes in the kernel to get it right. And then probably second is silicon carbide.

And not everyone just maybe uniformly, you know, maybe that's the way -

CONSULTANT KRESS: About how many nodes in that kernel? The kernel goes out to what? 0.2? That's the end of the kernel?

MR. ESMAILI: This is the kernel.

CONSULTANT KRESS: And about how many nodes do you use?

MR. ESMAILI: This is the buffer.

CONSULTANT KRESS: Yes, okay.

MR. ESMAILI: There's a big, big gradient that is continuing there. So, you have to put more nodes, as he says, near the surface of -

CONSULTANT KRESS: The kernel sort of gives the distribution?

MR. ESMAILI: The kernel is just to give you a contrast. To give you a contrast of whatever you -

CONSULTANT KRESS: But it's sort of a distribution measure also.

MR. PETTI: I think so.

MR. ESMAILI: And, you know -

MEMBER ARMIJO: Where is the silicon carbide on that? Is that part of the buffer?

MR. ESMAILI: Right outside here. It's this thin layer here. So, nothing is getting out.

MEMBER ARMIJO: Okay. So, that's really what's controlling the release -

MR. ESMAILI: That's the controlling. That's right.

MEMBER ARMIJO: Okay.

MR. ESMAILI: You look at the - like, for example, if you look at the same thing for strontium-90, you would see there is practically nothing in the outer layers. It would be just like a box.

What you see in the kernel is just like a box. So, everything is actually - you see that the retentive, you know, how much - how much gets - is still inside the kernel after two-and-a-half years. So, not much has gotten out.

And as a matter of fact on the bottom here, you see that, you know, like if you solve this in one to one-and-a-half year, you don't get anything outside of the fuel, you know. It just keeps going up until it builds up enough to get out. But, again, this is for an intact fuel.

Okay. So, once we do those normal acceleration, now we know the distribution of fission products inside the kernel, inside the matrix, everywhere. And so, we can do our transient calculations.

CONSULTANT KRESS: Is your expectation that when you get to the transport, the major component of mass and the moving around is the dust? It's overwhelming all the fission products.

MR. ESMAILI: I'll show you some results on the next - I don't know at this point. This is what we are doing right now. What we are doing right now, we are just doing a test calculation. And I'm just going to show you some results during normal operation with cesium.

But in this assumption, we just had cesium. All the particles were intact. Fission product is coming up as a vapor actually in the piping and -

CONSULTANT KRESS: So, there's dust in this.

MR. ESMAILI: So, there's dust in the system and it remains. In this calculation, it remains. But we had to make an assumption about the temperature of the structures, et cetera.

MR. PETTI: You know, the total mass coming out of this core is incredibly tiny. The partial pressures of these fission products are between ten to the minus ten and ten to the minus twenty atmospheres.

CONSULTANT KRESS: Yes, you get ten to the minus -

MR. PETTI: It's rare gas, yes, I mean -

CONSULTANT KRESS: I would expect the dust to overwhelm -

MR. PETTI: Absolutely. The dust will overwhelm, I think. And it's a timing issue though, you know, when the dust is, where it is relative to when the fission products get out, all those sorts of system issues. But also, you know, how things interact with surfaces.

I think a lot of the LWR models are wrong. I mean, they're talking about they have bulk phases. I mean, you've got enough cesium coming out of an LWR that you can -

CONSULTANT KRESS: They're huge.

MR. PETTI: It's a huge amount.

CONSULTANT KRESS: Yes, compared to this.

MR. PETTI: That is not the situation here.

MR. ESMAILI: No, we don't see -

MR. PETTI: So, it's a completely different physics in terms of where you are in terms of how things deposit and the like.

MR. ESMAILI: For these example problems, you're right. As a matter we don't see, you know, that the concentrations are so low that it never condenses into an aerosol form.

So, it just goes into circulating vapor and it just condenses on the surface as fission product. It's a different picture than LWR, but this is during normal operation.

CONSULTANT KRESS: So, just to give you an example, do your surfaces ever saturate or you don't have enough material to do that?

MR. ESMAILI: I don't think we have enough material to do that, no. You know, and talking about dust, we are talking about ABR, you know, three kilograms per year. So, even dust is not a very huge number, you know.

CONSULTANT KRESS: It's huge compared to the fission products.

MR. ESMAILI: It's huge compared to the fission products, but still a very, very low number, you know. So, it's going to be a very, very thin layer.

So, in order to test, you know, this, what do you call, normal operation mode, we already have a model for the core. You are seeing here, you know, the axial levels and radial rings in the core.

And we just - this is just a test. So, we are just adding certain control volumes to model the piping.

So, it goes out from the core and it goes around just to see how this normal axillary to steady state works. It doesn't mean - it doesn't intend to be a representative, you know, design which we are building right now, but this is for the test purposes right now.

So, here's a result for the PBR400 cesium distribution in the primary system. So, once we got the fission product distribution, now this is what gets released into the coolant.

So, what you see on the left-hand side, this is a circulating cesium mass. Okay. And after some time you see it comes to a steady state equilibrium condition.

So, whatever gets released, either its condensing or gets out, but whatever builds up on the heat structures, and on the right-hand side you see the cesium mass that's deposited on different heat structures, it just keeps going up.

So when I said, you know, we scale it, that means that we need to run this thing for a brief period of time, however that is, and come up with some equilibrium conditions. And the MELCOR, then we use that as a restart to do the accident calculation.

So, if I'm running this thing for about maybe 2,000 seconds, you know, again, these are like representative. But if I want to know how much it is going to be build up over ten years, it would just be an extrapolation of what it is.

CONSULTANT KRESS: Just a draw a line on -

MR. ESMAILI: It just goes up, right.

CONSULTANT KRESS: Yes.

MR. ESMAILI: We have to do a little bit differently for the nobles and, you know, that they don't deposit, but those, you know, decay. But that decay is already taken into account.

So, that's where we are. We are using the same models for the accident calculations also. And we are hoping to have the MELCOR - all the models in MELCOR by the end of this calendar year. The work is continuing on deposition and, you know, lift-off, et cetera, models. These are currently undergoing, you know, development.

CONSULTANT KRESS: Now, the meeting of protective action guidelines, maybe I'm asking the wrong person here, at the site boundary is, seems to me, like dominated by the normal releases.

When you depressurize and stuff, the normal stuff that you're looking at here that dominates that rather than additional releases; is that correct?

MR. PETTI: Yes.

CONSULTANT KRESS: So, you have to get this part right.

MR. PETTI: Yes, but that's not the hard part. That's pretty easy.

MEMBER REMPE: Did they do a COMEDIE test?

Did it finally go through? I know they talked about it, and so are you benchmarking against it for something like this?

MR. PETTI: Yes, there were a couple of COMEDIE tests done before that was shut down. So, they had done some additional benchmarking, but they would like to do more.

MEMBER REMPE: They would like what? I didn't hear.

MR. PETTI: To do more. They want us to set up a loop.

CHAIRMAN BLEY: Okay. Thank you very much.

MR. BASU: Chairman, I'm looking at the clock. What would you like to do?

CHAIRMAN BLEY: I'm expecting Joe to go through thirty slides in ten minutes. Do you think you can do that? Maybe we should have started at 8:00 this morning instead of 10:00.

Joe, you got two packages.

MR. KELLY: Yes.

CHAIRMAN BLEY: How about we do one of them before lunch, and one after, and we'll shorten lunch a little.

MR. KELLY: That sounds perfect.

And so since we've been talking about

computer codes, I'll continue on that line and go with the PARCS/AGREE presentation now.

Then after lunch, we'll do the supporting experimental programs. Okay?

CHAIRMAN BLEY: Yes.

MR. KELLY: And so just what I said, I'll be talking about the PARCS/AGREE development. And so now what we're talking about is core analysis.

Here's my infamous schematic again. And the point here is just to remind you that we use the PARCS/AGREE codes for both the normal operation steady state, and for reactivity insertion transients where we need a 3D kinetics capability.

So, PARCS is the NRC's core neutronics simulator. It's three-dimensional. It has both a cylindrical and hexagonal geometry. And it already had some preliminary validation work done for pebble-beds. So, we weren't starting from scratch there.

But to make it applicable to prismatic, what we're doing is implementing a triangle-based polynomial expansion method which they call TriPEN. And so, that's for prismatics. And I'll show you what that is.

We also have to improve the cross-section generation capability. And what we're talking about now is how you go from SCALE to PARCS. And in particular, implementing what are called assembly discontinuity factors. And I'll give you an idea.

They're correction factors that allow a homogeneous nodal code to match both the flux, which is over the node, and the currents, which are on the boundaries, match what you would get from a heterogeneous transport code.

And since I'm a thermo-hydraulics person and not a reactor physics person, that's about all I can say on that.

The other things that they're working on, well, for the future, they haven't started it yet, is the microscopic depletion capability. And this is needed so that you can follow a core over its three-year life.

And then we'll be participating in a new OECD benchmark that's being led by Idaho. And that's for the MHTGR-350 design. And so, that would be similar to what was done very successfully through the PBMR400.

So, this is what TriPEN looks like. So, you see a fuel element. And this fuel element not only has the coolant channels and fuel compacts in it, but there is a very large, almost four-inch diameter hole, which can be for the reserve shutdown system or for control rod, depending upon which fuel element design you're talking about. So, consequently there even within this one fuel assembly, there are a lot of heterogeneities.

So, what we're doing now instead of averaging the cross-sections over the entire fuel element or hexagon, it's breaking that hexagon into six triangular sectors and averaging the cross-sections over each sector. And then doing a nodal solution for each of these triangles and matching those up.

And so, what I'm going to show now is a benchmark for the HTTR. In this case, it's a numerical benchmark. So, the truth in this case is going to be a Monte Carlo transport code, which what we'll be using is KENO.

And what we'll be doing is comparing the nodal diffusion solutions of PARCS to the Monte Carlo transport solution.

So, we're going to use the HTTR geometry, that's the high-temperature engineering test reactor in Japan. It's an operating reactor. But we're going to simplify the geometry and we're going to look at a 2D plane.

So, Case 1 is very simple. It's five rings of fuel elements with uniform enrichment, and just one ring of a reflector.

Case 2 is four rings uniform enrichment. Two rings for the reflector.

Case 3 is now starting to look more like what the HTTR actually looks like, because they have reflector blocks sitting inside the core region and there are four different enrichments in this core at the mid-plane. So, those are shown in the middle of the slide.

Case 4 was added to be able to compute the control rod worth. And so in those reflector assemblies which are inside the core, you will now notice the open circles which are instrumentation guide tubes, and the dark blue closed circles which are the locations of the control rods.

And these are the results. So, I'm going to show results for both k-effective, and then the delta pcm for all four cases.

KENO is the Monte Carlo model that's built into the SCALE package. And so KENO-CE is a continuous energy version which Mourad talked about. And so for our purposes, that's the truth because we're doing numerical comparisons here.

KENO-MG is the same thing, but now breaking the cross-sections up into different energy groups.

So, by comparing the multi-group to the continuous energy and comparing those errors, and that would be this line where they're all in the order of negative two to 300, that's about as good as we could hope to do. This kind of defines, if you will, how good our codes have to be. So, if the PARCS code can get within those values, we're doing very good.

The next row is labeled "TRITON/PARCS." TRITON is the lattice physics code inside of the SCALE package. And so, that's what's used to actually compute the multi-group homogenized cross-sections that PARCS uses.

So, if you don't have the reflector ADFs, you get relatively large errors in the pcm. Unacceptably large. And this is what drove them to have to use the assembly discontinuity factors.

And when those were put in, we get very good comparisons here for all four cases, including the rodded case, but this isn't all of the story.

The fluxes in the thermal region are not as good as these numbers. So, there's more work to do, and that's what they're working on now.

AGREE, which stands for Advanced Gas Reactor Evaluator, is the thermo-fluids module of PARCS - although, actually, it's the other way around. AGREE becomes the main code, and PARCS becomes a library that it calls. And we did this having to do with configuration management.

Because if you were sitting in, say, the EPR or one of the BWR MELLA-plus presentations, you would hear about TRACE/PARCS. So in that case, TRACE is the driver of PARCS' two-phased flow code using PARCS as a library.

So, that's how we do the coupling to maintain the configuration control between different reactor types.

AGREE started out as a three-dimensional, two-temperature porous medium approach. It was developed to have a pebble-bed capability.

So if you will, it's a modern, rewritten version of the legacy THERMIX/DIRECKT codes. And so we took this and it already had some validation for pebble-bed, because that's what it was developed for, and we're extending it to prismatic.

The very first thing we did was extend it to model a prismatic using the existing r-theta-z geometry. And I'm going to show you some results from that, because we use that to review the scaling of the Oregon State high-temperature test facility. And so, that's an example I'll show.

But of course modeling a prismatic reactor in r-theta-z isn't good enough. So, what we're doing is putting a new prismatic capability in, and that has two parts. A bypass flow model, and a three-dimensional triangle-based heat transfer model. And I'll show you what both of those look like.

So, this is the bypass - the new bypass flow model. So, you see a drawing of the core on the left and on the right.

So, each fuel element is divided into six triangular sectors. Within each of those sectors, there would be one typical fuel compact, and one typical coolant channel.

The bypass channels are these new control volumes which are treated like subchannels. So, that models the gap between two fuel elements. And of course there are crossflow junctions between all of these subchannels.

At the axial interfaces where two fuel elements come together, due to irradiation damage you can open up gaps here.

And so, we have to also allow bypass flow or crossflows between a fuel element and a bypass gap at those interfaces, as well as potentially between the coolant channels going around a fuel element.

So, all of that numerical capability is now there, but we have to develop the constitutive models for it. And then as Mike asked earlier, we have to develop a module to describe the distortions.

MEMBER CORRADINI: Can you explain the arrows that are going around the clock inside the element?

I didn't understand that. I'm sorry.

MR. KELLY: Okay. If this is a fuel element -

MEMBER CORRADINI: Right.

MR. KELLY: -- each triangle will have one typical coolant channel, and one typical compact.

MEMBER CORRADINI: Right.

MR. KELLY: So, at the axial interfaces if they open up -

MEMBER CORRADINI: Oh, I got it. I got it. Okay. I got it now. Okay. Sorry. Thanks.

MR. KELLY: No problem.

So, for the heat transfer model, we have to model this at three scales. The micro-scale, which is on the order of a transient conduction in a coated fuel particle, that's important for reactivity insertion because it's the fuel temperature that matters.

For the meso-scale, and now we're talking about the heat transfer on the order of the fuel compact through the graphite matrix to the coolant channel, that's important for normal operation because it gives us our fuel temperatures and our moderator temperatures.

The macro-scale is now the heat conduction radiation either between these triangular sectors or from one fuel element to the other to get on out to the reactor vessel wall. And those are important in the conduction cooldown scenarios.

This is what the meso-scale model looks like. And this is similar to what I showed you for MELCOR earlier.

So in AGREE, we go ahead and model the fuel compact with a 1D radial conduction solution, but then we used a lumped parameter model for the graphite moderator. So, the graphite just has one temperature, and then we have to specify appropriate conductances between the surface of the fuel channel and the graphite, and then the graphite to the coolant channel.

And so, we've compared these to 2D finite element solutions and found a way to specify those conductances.

The macro-scale, and now what we're talking about, as I said, is the conduction paths between the triangles within a fuel element, and then going across the gaps to the adjacent fuel element. And here we have to model not just conduction, but convection to the flow and the bypass gap and radiation across that gap.

So, we needed a special treatment for that because of the radiation. We need to know the temperatures on those two surfaces. So, all of that's being worked on. Should be finished by the end of the summer.

Now, the example I'm going to show compares a calculation done for the MHTGR, which is the prismatic design of General Atomics, and the Oregon State test facility.

And what I'm going to do is a depressurized conduction cooldown. And the objective of this was to check the scaling of the thermal resistances in the core of an OSU facility.

Since that's a reduced-scale facility, we also had to scale the thermal conductivities. If we put something that look like graphite in it, the thermal conductivity would be way too high and we wouldn't get the right temperature rise over the core and the reflectors.

So, you subtract the scaling -

MEMBER ARMIJO: Before you do that, could you tell us a little bit about the OSU facility, what it is and -

MR. KELLY: Next presentation.

MEMBER ARMIJO: Okay.

MR. KELLY: Okay. Sorry.

It's, you know, chicken and egg kind of question. That's hard to do, but you'll see a little bit of it in just a second.

It's made to look like the MHTGR. Okay. So, at the time I did these calculations, the only thing that was available was the r-theta-z model. The cylindrical geometry extended to the PMR.

And so rather than try to compare two transient - conduct transient solutions, I did a quasi-steady approach taking the decay heat at about the time of peak power.

So, it's about half the percent decay heat, and do a steady state to generating this decay heat in the core. It has to go out through the outer reflector core barrel to the reactor pressure vessel. And then radiated convected from that to the RCCS.

And so, I'm going to compare MHTGR and the HTTF. This is what the MHTGR looks like. It's not a very clear picture.

And then on the right-hand side is the AGREE model of it. Remember, this is r-theta-z. So, I had something like 14 radial rings, 12 azimuthal sectors and 30 some axial nodes.

MEMBER CORRADINI: So, just so I'm clear, the MHTGR-350 is what NRC has partially reviewed back in the '80s?

MR. KELLY: It wasn't a 350 then, but yes.

MEMBER CORRADINI: Okay. Because there is an SER published on this, right?

MR. KELLY: Yes.

MEMBER CORRADINI: Okay.

MR. KELLY: It's very similar to that.

MEMBER CORRADINI: Okay.

MR. KELLY: And that's - I don't want to speak for DOE, but it looks like that's the direction the General Atomics design is evolving to.

MEMBER CORRADINI: Okay. That's fine. I just wanted to make sure I understood the connection.

MR. KELLY: Yes.

MEMBER CORRADINI: Thank you.

MR. KELLY: No problem.

And these are the results. So, on the left-hand side for the full-scale prototype, and the right-hand side the reduced-scale model, these are isotherms calculated at the core mid-plane.

The center region is at 1600, and these are at 100 degree C increments. And they look very much similar with the minor exception of a temperature drop between the core barrel and the reactor pressure vessel.

And that has to do - it's kind of hard to scale emissivities and you have too much surface area in a reduced-scale facility.

And this is an axial slice. And, again, they look very similar. There are few discrepancies. You'll notice these isotherms in the reduced-scale facility are too close.

At that time, their top reflector wasn't thick enough. They had a little scaling problem. The other one, you can't see the colors as well here. There's a slight color difference in the top here.

In the full-scale, there was a very small recirculating flow that didn't show up in the test facility. And that's something that we'll be looking at. And that's all.

CHAIRMAN BLEY: Anything from the Committee?

Okay. Well, if it's okay with all of you we're not creating a problem, I think we'll take our lunch break here and we'll meet back here -- we'll start promptly at one o'clock. Try to get back just a few minutes before that so we can start promptly at 1:00.

Okay. At this time, we will recess until one o'clock.

(Whereupon, the proceedings went off the record for a lunch recess at 12:17 p.m. and went back on the record at 1:01 p.m.)

A-F-T-E-R-N-O-O-N S-E-S-S-I-O-N

1:01 p.m.

CHAIRMAN BLEY: The meeting is back in session, and I'll give it back to Joe to continue where we were this morning.

MR. KELLY: Okay. I will start with the presentation on the experimental support for the NGNP evaluation model.

And I should make the point that what I'm going to be talking about are NRC experimental programs. I'm not going to be describing the entire DOE effort.

So, we expect that the majority of the experimental database will be provided by DOE and/or the applicant. And obviously the fuel qualification program is an example of that.

Another example would be the full -- almost full height RCCS experiment that's going to be done at Argonne National Laboratory.

So, those are programs that we would not replicate.

CONSULTANT KRESS: Let me ask you about the fuel qualification program.

MR. KELLY: Yes.

CONSULTANT KRESS: Will you not have to --

when they get ready to load the fuel into the reactor, will you not have to have the capability to say, yes, that fuel meets the standards we'd like it to have?

So, you would have to have some program of fuel qualification.

MR. KELLY: Yes, yes. There was - you're really asking the wrong person. And, Stu Rubin, if you want to chime in, but I know that they did a program with Oak Ridge to come up with how to monitor the manufacturing process for the fuel and for the QC program that that has to go through.

What I'm talking here is the AGR irradiations and the subsequent PIE testing. And Stu will actually talk about that some in his presentation.

CONSULTANT KRESS: Okay. I'll wait.

MR. KELLY: So, the objective of this presentation is to give you a very high-level, quick overview of the experimental program that the NRC is supporting.

And of course we will come back to you at another time and give you detailed presentations on each of these experiments. In 15 minutes, I don't have the time.

So, this is what we're talking about.

There are two integral tests. The high-temperature test facility at Oregon State University, and that's actually a joint DOE-NRC program.

Then there is a proposed OECD program at the high-temperature engineering test reactor in Japan. It's the LOFC program. So, that would be run by JAEA. That's pending, you know, signatures of all the OECD signatories.

Separate effects tests, we have several of them. There's a pebble-bed flow and heat transfer test at Texas A&M. There will be a prismatic core heat transfer also at Texas A&M. That hasn't started yet. So, I won't be addressing it today.

Both of those are part of the same cooperative research agreement that spawned the Oregon State high-temperature test facility. So, they were part of a package deal.

We also have some air ingress flow tests at Penn State. There was a cooperative research agreement with University of Wisconsin for the emissivity of vessel components. I'll show you an example of that. And a brand new program is on core bypass in the prismatic, and that's also at Texas A&M.

So, this is the Oregon State facility. It's an integral effects test. It's co-funded by DOE and the NRC. Actually, in this case, the NRC funds actually come from the Department of Energy through the Memorandum of Understanding.

So, we're running this program together. It's reduce scale. It's quarter height and quarter diameter. So, the L over D matches the prototype. So, that gives it a little bit more than six meters tall and almost two-meter diameter for the vessel.

The power level, the max we can go to is 2.2 megawatts and has to do with facility infrastructure.

It is a full-temperature facility. So, we plan to match both the inlet and outlet temperatures of the prototype. So, the outlet will be almost 700 degrees C. That's a mixed mean outlet temperature. Core outlet temperature would be higher.

It's a reduced-pressure facility. We'll only be able to go to eight bar. So, when we do a depressurization, we won't be doing the entire blow-down. We'll just be doing the tail end of the blow-down. And of course that's to keep costs manageable.

And we're going to - the initial core configuration is going to be prismatic based upon the MHTGR 350 design. That was a joint DOE-NRC decision. And that's what's shown inside the vessel here, is the prismatic core.

But the vessel itself and all of its supporting balance of plant, RCCS, et cetera, has been designed so that we can switch to a pebble-bed, you know, should the need arise in the future.

CONSULTANT KRESS: Is this nuclear power?

MR. KELLY: No.

CONSULTANT KRESS: It's electric?

MR. KELLY: Yes. Oh, yes.

(Laughter.)

MR. KELLY: I should say core simulator configuration, but - so, the HTTF was designed to model a depressurized conduction cooldown transient.

And of course in the beyond design basis space, you might be talking about a double-ended guillotine break of the annular hot gas duct.

But we also want to look at breaks that are much more probable. Things like CRDMs and instrumentation line breaks. So, we're designing the facility with different break locations and sizes in mind.

It's been scaled for four different phases. The depressurization, again that's only the tail end because it's only from eight bar.

Then the air ingress by lock exchange, and actually I'll be showing you a slide of that from the Penn State separate affects experiment later on.

Then after the air has flooded the lower plenum, it might thence progress through the core by molecular diffusion eventually leading to a natural circulation. So, all of this at the facility has been scaled for to some extent.

It will have a reactor cavity cooling system that's operational, but we're trying to provide a very well-known, well-defined boundary condition for the vessel.

So, in our case, we're going to be using forced cooling. So, we're not actually trying to mimic one of the potential designs. That instead will be the job of a DOE program at Argonne.

MEMBER ARMIJO: What is lock exchange? What happens there? What do you mean by air ingress by lock exchange?

I'm not familiar with that.

MR. KELLY: Yes, it's a - I think it's actually a civil engineering term. So, imagine a pipe separating two vessels. One vessel has very heavy liquid, say, water, and the other vessel is air.

Open up the pipe. You get a stratified countercurrent flow.

MEMBER ARMIJO: Okay.

MR. KELLY: Okay. You can think of that in two-phased flow space, but it looks almost the same as if it's air, which - cold air, which is your heavy, dense fluid, and hot helium, which is your light fluid.

MEMBER ARMIJO: Okay.

MR. KELLY: So, that's what's called the lock exchange mechanism.

MEMBER ARMIJO: Thank you.

MR. KELLY: And we've also looked at scaling facilities for both normal operation and pressurized conduction cooldown.

We can do a pretty good job of those if we switch simulant gases. So, instead of using helium, we use nitrogen. Then we can match the Reynolds number during normal operation pretty well, and also for the PCC.

Now, we can only do normal operation for the prismatic, and it has to do with being able to heat the pebble-bed since we're not generating heat within the pebbles with nuclear -

CONSULTANT KRESS: Will you do the air ingress at the high temperature? The 600 and -

MR. KELLY: We'll do it at prototypic

temperatures, yes. But we do - I should say our air in this case will be nitrogen.

CONSULTANT KRESS: Okay.

MR. KELLY: Okay. Because we'll be having graphite heater -

CONSULTANT KRESS: I was going to worry about your -

MR. KELLY: Our electrodes are going to be graphite. So, it will be helium nitrogen rather than helium air, just for that reason. We want to do more than one test.

CONSULTANT KRESS: Yes, that's what I was worried about.

MR. KELLY: I don't have a great picture of the core, but this gives you an idea. This is a unit cell. It's one of our fuel elements, if you will. There will be 66 of these. One for each of the fuel elements in the MHTGR design.

So, what you have is a hexagon. It's about three-and-a-half inches flat to flat. So, it's 1/16th the area of an actual fuel element.

There's a graphite electrode in the center with a little gas gap around it. There are also electrodes at the vertices, which means they get shared with the neighboring fuel element.

The white circles are the coolant channels. They're prototypic diameter, which is a good thing. That helps with the Reynolds numbers. And they end up giving us the, also, prototypic porosity. So, we're able to match both of those.

MEMBER ARMIJO: How does that work with a pebble-bed?

MR. KELLY: Well, we will be using a random - I mean, assuming we go that way at some point, it would look much more like the old SANA test facility in Germany or the HTTU facility that was in South Africa at Potchefstroom.

MEMBER ARMIJO: So, you'd use different heater elements that are -

MR. KELLY: We would have graphite electrodes probably buried in a central cylinder. And that central cylinder would then be surrounded by a packed pebble-bed.

MEMBER ARMIJO: Okay.

MR. KELLY: And then you conduct the heat radially outward. That's why you can't do normal operation because the temperature drop between that cylinder and the bed is too large at full-power type conditions.

What I haven't mentioned is actually the blue part in here is our, if you will, graphite moderator. But we can't use graphite in this because the thermal conductivity is too high. So, it's actually going to be a designer ceramic.

In order to get the right radial temperature drop during a conduction cooldown, we needed a material with a thermal conductivity 1/8th of the irradiated graphite. So, we're talking something on the order of four in SI units.

And so, we're actually building, you know, having a company make a specific ceramic mix for us and then casting it in a mold.

The next integral experiment is a potential one. It's pending signature. It's OECD HTTR-LOFC program. And I've already said before that HTTR stands for high-temperature engineering test reactor. Operated by JAEA.

It's an actual, real, operating test reactor. 30 megawatts. Prismatic blocks. And, you know, graphite moderated, helium cooled. And it has operated with an outlet temperature as high as 950 degrees C.

CONSULTANT KRESS: Just out of curiosity, what does LOFC stand for? Loss of fuel cooling or -

MR. KELLY: Loss of forced circulation.

CONSULTANT KRESS: Loss of forced circulation.

MR. KELLY: Okay. So, the program would have three tests. The initial conditions are shown here, but what's more important are the test procedures.

So, in all three of these tests, you start by tripping the gas circulators. So, there's no flow at all in the primary system. So, you lose all of your heat rejection capability. All your active heat rejection capability.

Then you do not scram the reactor. So, it's an ATWS as well. Instead the reactor is going to shut itself down with its negative temperature coefficient and then xenon.

For Test Case Number 3 in addition to those two faults, we're also going to trip - or they are going to trip the pumps on the vessel cooling system, because the RCCS through this plant is actually forced circulation as well.

So, by tripping those pumps, they turn off the RCCS. So, that would be a 100 percent failure of the RCCS system. And that's why Run Number 3 starts from a low power or low temperature, because we don't have the passive rejection.

And this is the expected results. This is reactor power versus time. And you will note the scale is focused in.

So, the blue curve, for example, started out at 30 megawatts, which of course is not shown here. And the red curve is the nine-megawatt case.

And so, what you see is the plant shuts itself down due to the negative temperature coefficient and xenon.

Then as the xenon decays away, you can go re-critical and come up to some new steady state power with a much hotter core.

And so, it's a very good test for something like PARCS/AGREE coupled, you know, a coupled reactor physics thermo-fluids.

MEMBER ABDEL-KHALIK: How does the MTC vary with burn-up?

MR. KELLY: I'm sorry, I don't - I can't answer that. I'm sorry. You're right that it does, but it's outside my area of expertise. I'm sorry.

MEMBER REMPE: How would the tests differ from what they did years ago at ABR? Because didn't they do a loss-of-flow transient at the ABR?

Of course it's a pebble-bed and I know that, and things like that, but how far did they go?

And was it just one test they did or was it multiple tests?

MR. KELLY: I don't know. We do have a staff member who actually worked at ABR. So, I can ask him.

MR. PETTI: Yes, I think these are very similar. ABR did more tests. HTTR did similar. This is just doing in a prismatic, largely, I think, as sort of -

MEMBER REMPE: Right. Okay.

MR. KELLY: And it also gives us the opportunity to be able to do post-test analyses with the experimenters in hand.

So, when you have a question about how you need to model something, you can get the answer which is sometimes all-important.

MEMBER REMPE: Of course.

MR. KELLY: Now, we'll talk about the separate affects test. And so, the first is the pebble-bed flow and heat transfer test at Texas A&M. It has four different areas of investigation.

The first was just simply pressure drop in a randomly-packed bed. We looked at both annular and circular beds because the design changed in the middle of this. That work is complete.

We also looked at the radial porosity distribution, and I'll show you an example of that. What they're doing now is looking at the radial velocity or, if you will, flow profile. And that's to try to quantify what the wall bypass effect is.

And then also a test they're starting now are convective heat transfer for the pebble. So, this is pebble-to-gas-heat transfer. So, not only is it a function of Reynolds numbers you normally see in a textbook, but it is affected by how close the pebble is to the wall. And all of these beds, especially if you go to an annular core, are relatively thin.

So, for the radial porosity distribution, what they did, they paired -- matched the index of refraction with laser-induced fluorescence.

So, basically the test section were completely transparent. You could look through the bed completely. Then they would shine a very thin laser sheet vertically aligned through the bed while putting a fluorescent tracer in the fluid.

So, what happens is now the spheres in that laser plane show up as dark spots, and the fluid is light.

So, using image capture and image analysis software, you can see how much of that cross-sectional area is solid, you know, and move the laser over, you know, and so on, and build up a map of the entire bed.

You can actually figure out where each pebble is in a randomly-packed bed. Which if you know when we try to do a CFD model later, it's kind of nice because you know where the pebbles are.

MEMBER ARMIJO: Yes. Very slick.

MR. KELLY: Now, what you can then also do is take that an integrate it in the axial direction and get the actual radial profile.

So, this is kind of nice because it was nondestructive, it was randomly-packed bed. This was a very small bed. So, you notice the wall effect on porosity persists to the center of the bed.

On a larger bed such as actually you would see in the HTR module design, you're about 14, 15-pebble diameters or something. So, that effect dies out typically after about five or six pebble diameters.

CONSULTANT KRESS: What's the unit on your porosity?

MR. KELLY: It's just zero to one.

CONSULTANT KRESS: I know, but what's it measuring?

MR. KELLY: Volume fraction.

CONSULTANT KRESS: The volume fraction of a given -

MR. KELLY: Of the fluid. The volume fraction of fluid, yes.

MEMBER CORRADINI: So, this is volume fraction of gas.

MR. KELLY: Yes. In this case, it was P-cymene, but, you know.

So, this is the slide for the air ingress test at Penn State University. And we talk about it as countercurrent stratified flow, but it can exist not only in a horizontal component, but also a vertical.

So, the idea of these tests were to study the geometric effects on the air-ingress flow rate during what we might call lock exchange looking at both the break orientation, the L over D effect on the pipe and the break geometry itself.

And we've already completed the scoping studies using water-brine, and that's what's shown here and I'll explain in a second. And we're just now starting scaled experiments with helium air.

So, the tank on the right is full of a relatively heavy brine solution. And the tank on the left is pure water.

So, the tank on the left is, if you will, the actual reactor vessel. And the tank on the right is the reactor cavity.

And so initially when you open up, there's a fast-acting shutter here and you open it up, the heavier brine flows along the bottom of the pipe while the less dense water comes countercurrent to it back.

At the intermediate stage, and this is a fairly high flow rate where the level of this mixture in what would in fact be the lower outlet plenum has reached the crossover duct, the flow rate slows down significantly.

And in what they call the final stage where that mixture reaches the top of the crossover duct, basically the flow goes to almost zero.

And this is the helium-air test facility. They're just starting these test now. There will be three different break locations.

On the left-hand side you see a location for a horizontal break. And actually there's a two - both flanges can have the break.

And on the right you see the vessel itself which is to model, if you will, the inlet plenum or upper plenum of a gas reactor with a break location on top. And there are multiple locations for thermal couples and oxygen sensors. So, all of this is to give us data for code validation.

The component emissivity test, that was done as part of a cooperative research agreement with University of Wisconsin. So, this is a system they built, you know, as an infrared irradiation system.

The samples were held in a silicon block. There was one black body and I think up to six samples.

The optics actually rotated so they go from the black body to all of the samples and back again and do this as a function of time. It wouldn't just measure it once.

There was a heater and thermocouples on the block so they could control the temperature to block. There was also gas lines so they could change the atmosphere so they could age a sample. And that's the results I'm going to show.

This is for SA 508, which is the candidate material for reactor vessel, and 316 Stainless, which is the candidate material for the core barrel. 316 is also what we're making the reactor vessel at the Oregon State test facility out of.

And if you look at the SA 508 at 500 degrees C in air, there are four curves here going from one hour to four hours.

I realize you can't see these, but the point of it is the emissivity over the entire wavelength is between 0.8 to a little bit greater than nine. And it ages very quickly.

There's not any, you know, definite aging between one and four hours. Once you get to one hour, it's there.

When you go to 1700 degrees in an air environment, it's pretty much over 0.9 the whole way. So, typically they use something like 0.84 in safety calculations. So, that would work just fine for 508.

For stainless, however, it's different. At 500 degrees, you're down at around 0.35. That's basically bare metal.

But if you then take it up to 700 degrees C in air which would be even a little high for the core barrel, it goes up to about 0.5 and it ages very quickly.

You notice there's no difference between one and five hours here. So, within one hour of exposure it had fully, you know, gotten as much corrosion on it, oxidation on it as it was going to get. So, this is somewhere between oxidized and bare metal.

This test just started - well, it's in the design phase for the facility right now. It's also a cooperative research agreement. So, it's the bypass flow test.

And we'll be using the mixed index of refraction coupled with particle image velocimetry to actually give us flow measurements, you know, both in the axial bypass channels and in the horizontal planes between these.

So, the whole facility is going to be made out of an acrylic and then using the P-cymene to make it transparent.

There will also be pressure drop measurements both in the bypass and in the coolant channels.

It will be three columns, three hexagonal columns, with 18 coolant channels. That's not the coolant pattern we're actually going to use. There are 18 coolant holes in each, and they'll be two blocks high.

We will have three different shroud configurations so we can move these blocks apart and vary that gap diameter from something like one millimeter up to five.

We can also separate these axially to open up those cross-flow gaps and make measurements as a function of that gap thickness. We will also have one of these machines so that we can do wedge-shaped gaps at various angles.

And so, this test and some of the legacy tests will give us the information that we need to get the loss coefficients to be able to model the bypass flow if we know the geometry or if we know an idealized geometry. And that's all.

CHAIRMAN BLEY: Okay.

MR. KELLY: Are there any questions?

CHAIRMAN BLEY: No. Thank you. Very slick.

MEMBER ARMIJO: I like that porosity technique.

MR. KELLY: So, I guess the next speaker is Stu. You're up.

CHAIRMAN BLEY: Okay.

MR. KELLY: Just let me get my memory stick out of here.

CHAIRMAN BLEY: So, were you showing us some things that weren't -

MR. KELLY: I had some backup slides, but I didn't get any questions that made me to go them.

CHAIRMAN BLEY: We slipped up. I'm sorry.

(Laughter.)

MR. KELLY: There's always next time.

CHAIRMAN BLEY: Next. Stu, your package has your slides, and then it's got some more from Hossein on MELCOR.

Is that all one presentation, Stu, or -

MR. RUBIN: I'm just going to be talking about fuel performance and fuel fission product release. I'm not sure I understand your question.

CHAIRMAN BLEY: Never mind. It's a clerical mistake.

MR. RUBIN: All right. My name is Stu Rubin. I'm the senior technical advisor in the Office of Research. And this afternoon I'm going to be talking about HTGR fuel performance and fuel fission product release.

And as Dave Petti likes to say, this is the sine qua non of the design. Without fuel and good fuel performance, you really don't have a good design.

Okay. With regard to the objectives for the presentation, there's really two major areas. The first was at the request of the subcommittee, you wanted to hear about an overview of the DOE's advanced gas reactor fuel development qualification program. And I'll be covering that at a fairly high level.

The second part, I'll be talking about the NRC'S HTGR fuels R&D that we're conducting to support the NGNP licensing review.

And in this area, there are a number of topics I'd like to cover. One of which is how the DOE fuels program is supporting our work, our R&D work in the fuels and accident modeling arena how we are developing a code that was developed by INL called PARFUME, it's a fuel performance code, and how we plan to continue development and use it in regulatory applications.

I'd like to go over in a little different perspective, the fuels modeling that we are putting into or have put into MELCOR for predicting fission product release during normal operation accidents.

Also, activities that we are conducting to increase the level of knowledge and know-how within the staff in the fuels arena.

And, additionally, we have a small cooperative research project that's intended to develop additional information on the transport of cesium through silicon carbide.

And finally, and we talked a little bit about this, a guidance document that we developed which will allow NRC inspectors to inspect fuel fabrication at a fuel fabrication facility for the NGNP.

Now, for those who may not be totally aware, there's obviously two kinds of fuel for HTGRs. There's spherical fuel elements and prismatic fuel elements as shown here.

The fabrication of either fuel type begins the same with the fabrication of a very tiny spherical fuel kernel which can be either uranium dioxide or uranium oxycarbide fuel kernels.

The kernels then from a fabrication point of view, are overcoated first with a porous puffer layer made of pyrolytic carbon. Then there's a dense pyrolytic carbon layer over that. A dense silicon carbide layer. And then a dense pyrolytic carbon layer overall.

And then in the manufacture of the prismatic box, you take those finished particles. You overcoat them with a graphite powder. And then you press them into cylindrical compacts which are about a half-inch long and two inches - excuse me. Half inch in diameter and about two inches long.

And then the compacts are stacked and inserted into fuel holes as was described earlier, within the prismatic fuel block.

For the pebble fuel, again the fuel particles, typically U02 particle kernels, are overcoated again and then pressed isostatically into a spherical fuel pebble which is the size of a tennis ball.

Now, moving to the advanced gas reactor fuel development qualification program, there are a number of important objectives to this program. First and foremost, to develop and qualify fuel that supports the NGNP safety case.

And what I mean by that is fuel that meets the fuel performance requirements and the fission product retention requirements of the NGNP reactor at its design operating conditions or service conditions and the NGNP accident conditions.

Second, the program is intended to reestablish within the US a capability to make high-quality UCO-coated fuel particles at an industrial scale. And they're pretty close to that.

Third, to demonstrate by irradiation testing and accident condition testing, that the fuel that they've made in fact meets the specified fuel operational performance requirements in terms of particle failure rates. And it also meets the accident condition performance requirements in terms of particle failure rates.

Fourth is to increase the science and the basic technical knowledge of the phenomena of the effect, fuel product behavior and fission product release so that the models that are developed will be improved.

And finally, a lot of what we're talking about here is the last item, to develop experimental data that will be needed to model and validate the codes that will predict fuel failure and also predict fission product release in the safety analysis.

Now, this diagram all in one place talks about the major program elements in the AGR fuel development qualification program.

The first element at the top is fuel fabrication, as I mentioned, or fuel supply development. And the objectives of this element is to develop a fuel particle fabrication process that meets the fuel quality and fuel performance requirements demanded of the NGNP safety case.

And the strategy that's being implemented by DOE for this element is to first come up with a process that meets the design specs for the particle, but simulates and follows the German particle coating process to the extent possible to end up with the same properties, coating properties, and ultimately the same performance that the Germans saw in their particles, which was a very high level of performance in the reactors in Germany.

Next element, again, is fuel irradiation. That's going to be conducted at the design conditions. And in some case, in margin testing beyond the design conditions.

And the objectives of this element are first to develop data on irradiation fuel performance of the particles to support fabrication by feeding back the experience of how the particles perform into the fabrication process.

Also, obviously, they'll be part of the fuel qualification program, and also developing the fabrication process, and also to support the development of data for developing models and validating codes to use in the safety analysis.

Next element is post-irradiation examination and accident condition testing. And here, the objective is to again obtain data on fuel performance at the NGNP's design operating conditions, and also in the accident conditions, by actually looking at the fuel and taking measurements of various sorts which I'll get into.

Next element is fuel behavior modeling or fuel performance modeling. And what we're talking about here is actual particle failure prediction modeling.

And what it does is in its objective, is to model the structural and thermal and physiochemical behavior of the processes within the coated fuel particle with the objective of being able to actually predict when a particle will fail.

And the final element is the development of data to support fission product transport modeling for the safety analysis and the development of accident source terms. This needs to be done for both normal operations, as well as for accident conditions.

And I would mention that DOE and INL are also partnering with Oak Ridge National Laboratory, B&W Lynchburg and General Atomics as participants in this program.

Now, what this slide represents all in one place is an overview of the AGR fuel program activities here in one slide. The program consists of eight different fuel irradiations; AGR 1 through AGR 8, which is shown in the orange boxes.

Following each irradiation is fuel safety testing which is conducted along with post-irradiation examinations. And these are shown in the red boxes.

Each of the eight irradiations has a different purpose. And these are shown in the yellow boxes. And I'd quickly like to go through each of those.

AGR 1 involves fuel compacts with coated fuel particles with UCO kernels. And the coating layers were made in a laboratory scale using a German-type coating process.

The fundamental purpose INL would say for this irradiation was really to check out the irradiation capsule and train design which is going to be used in the later irradiations for fuel qualification, but also they included particles with slight variance on the coatings to try to see if there were any differences in performance to optimize the one they would pick for the actual production fuel.

Now, AGR 2 is what could be called a fuel performance demonstration irradiation. It involves a UCO kernel, as well as U02 fuel particles made on the industrial scale here in the US with industrial scale coater. But it also involves particles - excuse me - yes, U02 fuel particles that were made by CEA in France, as well as U02 particles that were made by PBMR in South Africa. So, those are actually being irradiated as we speak.

Now, the purpose of AGR 3 and 4 is to obtain data to develop the fission product diffusion coefficients in the various elements or constituents of the fuel, the coatings, the kernel, the matrix, et cetera.

And to do this, they - it's got to involve compacts with fuel that is designed to fail. And these particles will put off a well-defined, known quantity of fission products so they can then use that knowledge in analyzing what the diffusion coefficients would need to be, let's say, in the matrix to - once they see what the fission product concentrations are at the end of the irradiation via PIE.

Now, AGR 5 and 6, these are the formal fuel qualification tests. What these tests are, and there are two trains, 5 and 6, simply because they need enough particles to meet the confidence level on the statistic on particle failure rates when the irradiation is over.

And they will use production scale methods for making the kernels, the coatings and the compacts with the prototype fuel in it to the specifications. And then they will run it at the design conditions of the NGNP, the design service conditions in terms of, let's say, packing fraction, temperature burn-up, fast fluence and the like.

And from that at the end of the test, they will have a count on how many particles have failed out of the three hundred to 400,000 that are going to be irradiated.

And through statistics, they'll be able to infer what the design level failure rate would have been given a certain number.

Hopefully they'll have no failure rates, but they'll be able to check it. And that will be the basis for qualifying - saying that the fuel made to this spec on an industrial scale with these processes, give this level of fuel performance.

MEMBER ARMIJO: Now, that's under normal operating conditions, right?

MR. RUBIN: This is fuel qualification.

But the second piece to that then is to take those irradiated compacts and put them into an autoclave and heat them up in an accident condition simulation test. And then get data on particle failure rates under the accident condition.

So, that's the second part of the fuel qualification. And it's both the normal operation, as well as the accident condition testing for fuel qualification.

Now, the purpose of 7 and 8 is to obtain data to validate the fission product transport codes, as well as the particle failure rate codes, by running experiment to actually fail particles intentionally.

And so, you would have data to then see if your codes would be able to predict the number of particles that failed and when they failed.

And, also, to have data on the actual fission product releases to actually validate the codes that account for both failed and intact particles in their analysis, as does the code they developed, PARFUME.

Now, the last piece is something called fission product transport and retention. These are intended to be out of pile. They're designed to obtain data for developing and validating the models for fission product transport outside the fuel form that is within the helium pressure boundary, things like lift-off, plate-out, absorption and the like, and also tests to develop data for fission product transport in the reactor building. So, this is part of the fuel program as well.

Now, where are we in terms of the status of this testing, this work? This slide kind of gives you the status.

AGR 1 irradiation, that was completed in November of 2009. None of the particles out of the more than 300,000 that were irradiated were found to have failed.

The peak time average fuel temperature for this irradiation was at 1250 degrees C. Fast fluence was four times ten to the 25th neutrons per meter squared. And the burn-up was almost 20 percent of FIMA. And about 35 percent actually achieved the burn-up goal of 18 percent FIMA.

The PIEs, this fuel has been taken out of the AGR reactor and is now undergoing PIE. And they'll do things like measure fission product releases by examining the fission product content of the holders and of the matrix material.

Again, we're looking at the silicon carbide condition analyzing fission product, the inventories and taking a look at the micro structures of kernel and the conditions of coatings and so forth.

MEMBER CORRADINI: There's one thing you said there.

So, the 20 percent is 20 percent of the fissile inventory?

MR. RUBIN: Fraction of initial metal atoms that were fissioned.

MEMBER CORRADINI: Okay.

MR. PETTI: A hundred percent U235 consumption.

MEMBER CORRADINI: Oh.

MR. PETTI: Yes.

MEMBER CORRADINI: So, you burned out -

MR. PETTI: You've burned it out. We're on fumes, yes.

MEMBER CORRADINI: Okay. That's what I didn't understand. Okay. Thank you.

MR. RUBIN: Okay. You know, later this year they're getting ready for their accident heat-up testing. They've kind of gone through a shakedown. And they plan to actually start that heat-up testing on the AGR 1 irradiated fuel by the end of the year, as I understand it.

AGR 2, that's currently underway. And the detailed planning is underway as well for the follow-on PIE work and safety testing.

As far as AGR 3 and 4, that's really the heart of the fission product transport data acquisition work. The fabrication of that fuel is underway and the design of the capsules and the trains are almost complete. And the irradiations will begin as soon as AGR 2 is finished.

Turns out that because of the feedback that they expect to get from these irradiations to come up with a final design for the fuel qualification tests for 5 and 6, they're not ready to really get started with the fuel design work just yet until they see what the current fuel performance looks like from PIE, et cetera.

When you look at the schedule which is not shown here, it's a very, very aggressive schedule. And it needs to be so because it turns out that the fuel is on a critical path of the entire NGNP R&D program.

I'd like to talk a little bit in the way of segue into our - NRC's program about a code, a fuel performance code, a particle failure prediction code that I know has been developing for probably ten years by now. And it has been developed to support the AGR program to help them with their design and development and testing prediction activities.

They call the code PARFUME, and that stands for particle fuel model. And one would consider it really a state-of-the-art TRISO-coated particle fuel performance analysis code.

What it does is it calculates the failure probability of coated particles within the fuel element. And it also calculates the fission product release from the fuel element for both normal operation, as well as accident conditions, by accounting for both failed particles, as well as intact particles within the fuel form.

It models all the important phenomena, I would say, that can lead to particle failure, and these include phenomena such as internal pressure within the coated particles, cracking or de-bonding of the pyrolytic carbon layer away from the silicon carbide layer which can cause stress risers in the silicon carbide layer, migration of the fuel kernel toward the particle coatings, increased local coating stresses due to amount of roundness of the particle and silicon carbide layer thinning caused by fission product interactions.

It also handles the statistical variations in particle coating properties, including geometrical properties like silicon carbide layer of thickness, as well as mechanical physical properties.

So, because particles are made with variations from one to the next, the coat is able to take that statistical distribution from manufacturing spec and put it into code, and that's factored into the statistics of particle probability.

CHAIRMAN BLEY: Stu, I'm just wondering what kind of documentation there is on this to look at.

When I look at the photographs I see and the micrographs blown up like the one you showed on Page 3, these things seem extremely irregular the way they're grown. And modeling this seems like it would be pretty darn tough to do, and do well.

MR. RUBIN: And this is at a 30,000 foot level. I had another slide to describe what documentation exists.

There is a theory manual for PARFUME that we have received. There is a users manual that we have received.

We have the code. We have runs. We have the reference documents that the code used in developing its models and the data that was used and currently in the models.

So, there is a lot of data really - or reference material really in large part driven by our interest in acquiring the code.

So, I have a slide that I can make available to you that we have.

CHAIRMAN BLEY: Yes, I'd like that.

MR. RUBIN: We have it in-house.

CHAIRMAN BLEY: They may ask to see some of those if you have them.

MR. RUBIN: And we also have, and I'll mention this, we did receive training, a five-day training course on PARFUME, which got into a lot of detail on the coating and the numerical methods, et cetera, that are used to solve the base of the equations in there.

MEMBER CORRADINI: But to get to his original question --

MR. RUBIN: Yes.

MEMBER CORRADINI: -- I'm not sure if it falls under another part of INL than the salmon-shirted gentlemen over there.

(Laughter.)

MEMBER CORRADINI: Peach. Salmon.

What I guess Dennis was asking, and I was wondering about the answer, so there is a distribution function to how these things are made, right? Because there's a -- I would say there's a recipe.

You go make the recipe and then you inspect on some sort of statistical basis, what comes out.

So, this is essentially a deterministic

model that then has what I'll call a distribution of what could be the geometry?

I'm trying to understand the probability part of this.

MR. PETTI: What the code does is based on the QC data, you get a mean and a standard deviation for the population -

MEMBER CORRADINI: Okay.

MR. PETTI: -- for 30, 40 attributes. We measure all sorts of stuff. That's all in the code. The code then does - can do three different things, but it basically can do a Monte Carlo sampling, sort of what it does.

MEMBER CORRADINI: But it's a deterministic calculation with essentially a distribution function of all the key attributes.

MR. PETTI: Yes.

MEMBER CORRADINI: Okay.

MR. RUBIN: And there's a finite element model.

MR. PETTI: Yes, there's a finite element model and -

MR. RUBIN: So, you could have, you know, bends in the coatings, and the finite element will model that and pick up the stress risers associated with those bends.

MEMBER CORRADINI: Okay. But to the extent that I want to make sure I understand it is the reason you say the probabilistic is since I had the distribution function, that would affect how it ends up the peak pressures, thinning, diffusion, dah-dah, dah-dah.

MR. PETTI: Right. And then we use a Weibull failure, and that's how you get a failure probability.

MEMBER ARMIJO: Do you have a list of all the important attributes that you actually would measure in your fuel fabrication facility -

MR. RUBIN: Yes.

MEMBER ARMIJO: -- from the buffer thickness to the silicon carbide thickness?

MR. RUBIN: There are 77 specifications that were measured.

MEMBER ARMIJO: And then you'll have a process for measuring those?

MR. RUBIN: Yes.

MEMBER ARMIJO: Some experimental quality control process?

MR. RUBIN: Yes, yes. In fact, my very last slide is -

MEMBER ARMIJO: I'd really be interested in that because I come from a, you know, from Light Water Reactor fuel background. And, literally, our smallest thing we have to worry about is a fuel pellet.

Okay. But your fuel pellet and your cladding, you know, these tiny, tiny, little things in the hundreds and hundreds of thousands -

MR. PETTI: We have the means to measure all of that.

MEMBER ARMIJO: At some point in the future, Mr. Chairman, I'd like to get a manufacturing presentation on how this is done and -

CHAIRMAN BLEY: And I think there's a technical report associated with that kind of thing.

MR. RUBIN: Yes, we have a number of technical reports, but I'm sure we could make it available.

MEMBER ARMIJO: To me, a big concern I have is this obviously pretty forgiving process, but batch-to-batch variability on -

MR. PETTI: It's actually not as large as you think, but we're actually tackling some of those issues right now.

MEMBER ARMIJO: Okay. At some point, you know, we can get that in to the Committee.

MR. RUBIN: Yes, one of the goals of the manufacturing process is to basically meet the spec every time so you don't have any kind of surprises coming your way.

MR. PETTI: And Stu doesn't - I don't know if he has any slides on this, but what we're doing today in 2011 is just a hell of a lot better than what was done historically in 1975.

Okay. It's night and day. For those that are on the NEAC side, you'll hear about it in a few weeks.

But basically measurement science and our ability to control the process is just much better than it was in 1975. It's a real success story, I think.

CHAIRMAN BLEY: Yes, we would be very interested in that.

MR. RUBIN: My last slide, the last line is a good document for you to read on the area of your interest.

This next slide really covers the continuity equations in PARFUME. And you can see it covers thermal analysis, the stress-strain analysis, fission product transport analysis in the particles themselves, as well as within the fuel form whether it be a compact or a pebble matrix. And it also handles intact and failed particles within there as sources of release within the fuel form. And then release from the element itself.

The reason I put up this slide in two colors is that MELCOR has these very same models as the ones shown in black. Where MELCOR and PARFUME differ is in what's shown in read.

PARFUME actually does the stress-strain calculation and calculates the peak stresses, and then compares it to a Weibull failure probability curve as a function of stress to come up with a failure probability for particles that are at that stress.

PARFUME doesn't do that. What we're doing - excuse me. MELCOR doesn't do that. With MELCOR, we're putting in an input-defined failure rate in the form of a failure fraction curve or failure fraction response service, or we could put in something different.

The main reason is it's a very time-consuming type process to do this and it's just not suitable to have this kind of a failure predictive model in MELCOR.

MEMBER REMPE: And the impact will be, I mean, if you compare the two, what we'll get?

MR. RUBIN: Yes, I'll talk about that -

MEMBER REMPE: I'm sorry. Okay.

MR. RUBIN: -- as we go.

This next slide is really how are we going to continue to work really in parallel with INL on future development of the code? And what this involves, really, is summarized in this slide.

The slide indicates what kind of parameters we target for updating or evaluating, what the source of the data is, when it's available and when it's going to - the effort's going to be completed. And as you can see, the entire effort is really tied to the AGR irradiation test PIE programs.

The first row in the AGR program, I didn't mention this before, is to develop data on the mechanical and physical and physiochemical properties of the fuel constituents. The silicon carbide layer, the kernel, et cetera.

And these properties will make the code more specific to the actual materials and the properties of materials of the particles that DOE is making rather than taking data from, let's say, the German coating process experience and so forth.

So, with those updates, the ability to predict stresses and strains and failures in a code will be more precise and accurate to the actual fuel that it's trying to calculate failures for.

And then beyond that there's a number of benchmarks that we plan to do pre-test and post-test predictions of the different fuel irradiations, AGR 1 and 2.

By the time you get to AGR 3 and 4, you're actually developing data for modeling of the diffusion coefficients for the fission product release calculations.

And there, the post-test prediction or benchmark would actually rerun the calculation for the actual test conditions with the new diffusion coefficients to see how close you get to predicting the actual releases that were measured by the PIEs, et cetera, in the post-irradiation, also post-accident condition testing.

And then there is the AGR 5 and 6 which, again, will be the - used to do qualification testing. And then 7 and 8 is validation testing where they actually go into failed particles and you'll be able to see if you can predict those particle failure rates with PARFUME.

Now, with regard to MELCOR, we're not going to be able to predict those failure rates.

We're going to have failure fraction curves or response surfaces.

What this test will do, it will allow us to see if those failure fraction prediction curves are wildly conservative or pretty good for the purpose we had set out to have an input database rather than a model to actually predict the failure rate. So, we use it for MELCOR as well.

MEMBER ARMIJO: Stu, before you go on, just to make sure I understand what constitutes failure in a particle, is it simply a through-wall crack through the silicon carbide layer? Is that -

MR. RUBIN: It's the failure of all layers. It's the failure of all layers. You start with a particle that has all the layers, silicon carbide, the two pyrolytic carbon layers perfectly intact, and then a failure of a particle which is observed by a large increase in gaseous releases. You know you failed all the layers at that point.

Now, the layers don't disappear. There's a crack that allows that gaseous release. And then of course there is the metallic release that will come through that crack as well.

The way that codes typically model that failure is at that point of failure, the assumption is the layers just disappear and you now have a kernel. And the kernel now, and the releases from the kernel into the matrix are what you're now modeling in terms of integrated release from that pebble or compact.

MEMBER ARMIJO: So, if you have a failed silicon carbide layer, but the buffer and the pyrolytic graphite layers over a particle are still okay -

MR. RUBIN: Yes.

MEMBER ARMIJO: -- you won't have much release, or you will?

MR. RUBIN: Well, then you have a variant on - I wouldn't call it particle failure. It would be a failure or a defect in the silicon carbide layer.

And there, the modeling is the kernel and the two pyrolytic carbon layer, but no credit for the silicon carbide layer.

In fact, if you look at the specs for manufacture, there is a spec on defective silicon carbide layer from manufacture.

So, that fraction of particles should be built into your code for fission product, core-wide fission product release.

MEMBER ARMIJO: Well, I had the misconception that the silicon carbide layer was effectively the cladding.

And if you lose that -

MR. RUBIN: It's the most important layer.

MEMBER ARMIJO: It is the most important one still.

MR. RUBIN: Yes.

MEMBER ARMIJO: And so, failure of that silicon carbide layer would constitute near failure of the whole particle -

MR. RUBIN: Well -

MEMBER ARMIJO: -- even if the others were okay?

MR. RUBIN: If you look at the categories of particles, you have intact particles where all the layers are intact, you have particles that can be defective for manufacture where the silicon carbide layer is missing or cracked, and that has to be modeled as defective from Day One -

MEMBER ARMIJO: Okay.

MR. RUBIN: -- and then you have particles that fail under irradiation or accident conditions, and those failures are essentially failures of all the layers.

We don't have a model that says, oh, only the silicon carbide layer failed at this point. It's all the layers fail at that point.

The reason is that it may happen, and I don't know that I've seen it from PIEs, but we'd have no real way of detecting that it happened.

MEMBER ARMIJO: You probably can't distinguish, you know, something is coming -

MR. RUBIN: You can't distinguish because there's no -

MEMBER ARMIJO: Fission products are coming out.

MR. RUBIN: -- there's no fission gases released to tell you that that happened.

MR. PETTI: That's why we check. That's why we check in PIE. We check how much cesium is outside of the silicon carbide.

We take these compacts and we de-consolidate them. And we measure the cesium content in the fuel matrix. And if you had defective particles - defective silicon carbide, the cesium will get through. The noble gases will not be measurable online because pyrocarbon holds in the noble gases.

And so when you look, for instance, at the entire German experience, there was no silicon carbide failure under irradiation.

What was measured in the PIEs was

completely consistent with what was measured in the QC that you would expect a certain number of defects. And they looked at it and statistically it all, you know, made sense.

And, again, most of the - the silicon carbide's in compression. So, as a ceramic, that's the way it's designed. It's not supposed to be intention if it's intention than you run the risk of failure.

MR. RUBIN: But interestingly if you look at the signature of, let's say, krypton release when the whole particle fails, you don't get a signature necessarily that is all the layers disappeared. You get something less than that.

Okay. But the modeling for reason of conservatism, just assumes that all the layers are gone at that point.

Okay. What are we going to do with PARFUME? We're not going to use PARFUME as part of our evaluation model that Joe Kelly talked about.

It's a very powerful tool. But because of the long run times, it's just impractical to use it for the hundred or more nodes of PARFUME that MELCOR would need to do a calculation of particle failure rate at any given time.

What we do plan to do with PARFUME is to evaluate the applicant's failure prediction model that they will be using in their safety analysis codes.

We also will use it as DOE, and INL use it, to do sensitivity studies to assess the affects of variations in the geometry and the properties, in the irradiation conditions and its affect on failure probability.

We also plan to use it to help benchmark MELCOR. Because, as I said, both codes really have the same fission product transport model in them, we can use one code to benchmark against the other in a code-to-code type benchmark not only of kernels, but also of actual particles in compacts by modeling the actual experiment. And I'll talk a little more about that later.

It's also a very good tool for training the staff on the underlying principles of particle behavior, failure and fission product release. And we'll be using it for that.

And we can use it to assess the effects of perhaps some testing, startup and safety testing that may be envisioned for NGNP to satisfy ourselves that there's no harm to be done in doing these tests in terms of greater performance of the fuel in the core during operations.

And as a tool to get insights into what potential administrative limits or - on the fuel temperature or on the burn-up that may have on the source term.

As compensatory measures, there's some thought that we might need to have compensatory measures to have some limits on the conditions of the core, temperature burn-up, et cetera, to assure that we're well within the safety analysis. And PARFUME will give us some insight as to what those benefits could be.

Just a quick slide that talks a little bit about the differences in fission product transport modeling between prismatic fuel elements and pebble fuel elements. For the most part, they are identical.

I would point out the asterisked item where I speak about diffusion through each of the layer of coatings. The asterisk item would say if you have failed particle, you don't actually model or give credit for the diffusion or holdup of fission product transport in those failed particles through the layers.

Some of the differences that come up is there's a gap between the outside of the compact and the inside of the graphite block in the fuel hole. And that little gap provides some additional resistance. And there is a solid-to-gas and gas-to-solid transport process. And that needs to be modeled.

And of course there's the whole graphite block in a prismatic block that needs to be modeled in terms of fission product transport in a prismatic fuel element that doesn't get modeled in a pebble fuel element. For the most part, they're really the same modeling needs.

Okay. What this slide does is it takes the earlier slide with PARFUME and it really talks about, well, what can we use out of the AGR tests in terms of actual experimental data to help us benchmark not only PARFUME, but MELCOR in terms of fission product transport. This would be Code 2 data-type benchmarks.

And really, all of the AGR experiments can be used for that purpose. Because at the end of the day, we will have a well-defined geometry, well-defined fuel, well-defined irradiation conditions and very good data on fission product release during the irradiation not only in terms of gaseous release online measurements, but also in terms of PIEs of actual fission product inventories outside the fuel particles, outside the fuel compacts.

And we can take that data to then see if we could predict that with not only PARFUME, but also with MELCOR.

So, the thought here is that we will use all of these irradiations to actually do code-to-data benchmarks from MELCOR.

The difference lies in the actual code or fuel particle performance aspect of these tests where particles fail.

Because MELCOR doesn't have a fuel failure reduction model other than that response service, we will use that aspect of these tests where particles do fail, in some cases, by design in 7 and 8, to benchmark PARFUME to see if it prediction abilities for predicting particle failure rates is good or otherwise.

And for MELCOR, we can use to see if whether or not that response service is doing a good job for how many particles have failed under the conditions that the test was run at.

So, it's a very simple geometry, but it is what I would call a unit cell. It is a stack of compacts with particles in it. You have coolant flow around it and ability to monitor fission product releases and distributions very well.

Okay. A question came up by the subcommittee, well, how are the lessons learned and results from the AGR program providing insights or help to the NRC's activities?

And this slide is intended to list a number of areas where DOE's AGR program has given us benefit.

For example, that first item we had a number of questions in the early going on what was important to consider in modeling fission product release from, let's say, prismatic fuel design and what was not so important.

Things that came up like do you need to model thermal diffusion as a separate kind of diffusion category in addition to Fickian diffusion which is simply concentration-based?

And there's evidence that there are some conditions where you could have a separate thermal diffusion affect.

And in discussing it with INL, we kind of were agreed that it may not be considered that it will be effectively part of the effective diffusion coefficient in a Fickian-type diffusion approach.

How to model the gap between the outer surface of the compact and the inner surface of the fuel hole, how to model that fission product barrier, if you will, we got some ideas on that.

A number of questions came up on temperature profiles in the particle, in the compact. And it was their experience with PARFUME, our experience with the way they - the data they got from that test was very helpful. So, I would say that was a very big success for us in obtaining help from them on that.

Obviously, the AGR fuel irradiation fission product release data that I talked about before, that's starting to come to us now and will continue to come to us and will benefit us in benchmarking MELCOR.

We already got from them PARFUME's prediction from the IAEA CRP-6 benchmarks. These were code-to-code. In some cases, code-to-data. And so it's - and I think that Hossein talked about his initial prediction using the MELCOR model against the PARFUME or INL results. They were very good results.

One could argue that perhaps neither is accurate, but we have another code-to-code database that they provided to us, and there are other things listed there.

So, I would say this is a big area of success. I view it as kind of a unique opportunity for research to probe and poke a quasi designer organization on how they model important phenomena in the absence of having a real design organization that's made an application. So, this is good for us.

I'd like to turn now to a small project we had with the University of Wisconsin and Todd Allen. We had a cooperative agreement with them to improve our understanding of fission product cesium transport in silicon carbide.

The goal there was to get a better understanding of what are the key properties or characteristics of silicon carbide that have a controlling influence on the rate of cesium transport in silicon carbide simply because everyone is using for now at least, the old German data on their diffusion rates and claiming that while it applies equally to us, we want to see if there's any basis for saying that that would be a good bet other than just kind of looking at the micro structure and looking at how the grain size and orientation of grains look and say, yes, I guess you could reference that data.

So, this was intended to give us some

deeper understanding of the processes of cesium transport in silicon carbide.

Basically, the scope involved bulk diffusion and grain boundary diffusion. We want to look at the effects of micro structure, as well as temperature.

And the investigation involved both molecular modeling methods, multi-scale methods, atomic at the grain level, and then at the entire - level of the entire composite.

And it also involved measurements of actual cesium diffusion on irradiated silicon carbide-wide diffusion couples.

Now, in terms of what came out of this work, that's shown on this slide, basically. In fitting the diffusion couple data to analytical models, it yielded a bulk diffusion coefficient of about ten to the minus 18 atoms per meter squared per second at 1200, which is really in a good range for what is typically measured.

The atomic level calculations indicated a bulk diffusion involving what they would call, and I don't totally understand this, charged defect clusters having an energy, activation energy of 5.5 eV. And those are shown as Lines 2 and 3, that kind of cluster, on this slide.

What they found, though, that the predicted solubility of cesium and silicon carbide was, by their calculations, very low and was not consistent with measured release data or their diffusion couple experiments.

And they could not understand that, and so they postulate or hypothesize that there were other more complex mechanisms. And the proposal was that there could be some impurities that were acting to affect the solubility and diffusion.

Well, there could be some affect of a neutron irradiation field in terms of creating vacancies or defects in the lattice that would speed up the process.

And so, they proposed a follow-on research to look at those effects. And we have approved the latter to investigate the affect of neutron irradiation in enhancing the absorption of cesium and the diffusion of cesium through the creation of defects in the lattice.

It is a particularly important question because it turns out that historically the data on diffusion coefficients have been taken from post-irradiation theta testing after irradiation is over.

And so, the hypothesis would say if proven true, that that might not be a conservative way to get to diffusion coefficients.

MEMBER ARMIJO: You want to be generating your -

MR. RUBIN: You want to be generating -

MEMBER ARMIJO: -- at the same time.

MR. RUBIN: At the same time. So, that's why we're interested in seeing how that may turn out. We're not sure.

Okay. The last item is basically this question of oversight and inspection of fuel fabrication facilities.

We all would agree that how you make the fuel and making it consistently to spec is the key to fuel performance in the reactor.

And so, the idea was let's ask the people who have all this experience in developing these methods for DOE and that program, to write up a - what's important and how you would go about examining whether or not those important characteristics and processes are being conducted within - in the right way.

So, they turned around and took everything that they had learned, and this is Oak Ridge in the lab scale work to develop kernels and coatings and compacts, and asked them could you please put together this document that describes what's everything that's important and how would you go about inspecting for and sampling of key important areas?

And they did that, an exceptionally good job. It's a tutorial on the one hand, but it's also very insightful on the details of how to go about doing this.

And it was captured in the documents shown at the bottom. If you have an interest in really understanding how they do quality control and what are the key parameters, as Dave mentioned, this document covers it all.

Now, it would be accurate to say it does represent a picture of a lab scale operation, but many of the processes are scaled up for industrial scale. But most of, I would like to think, of what's important is still going to be retained even at the scale-up level.

Okay. And that kind of covers everything I wanted to cover.

MEMBER ARMIJO: I'd like to get a copy of that report.

MR. RUBIN: Yes, it's very good.

CONSULTANT KRESS: Stu, do you envision NRC actually showing any oversight over the fabrication facility at all?

MR. RUBIN: Well, I -

CONSULTANT KRESS: You don't know.

MR. RUBIN: We have people go out to Light Water Reactor inspection facilities. And the areas of interest are contamination control, criticality control and things of that sort.

Here, the interest is in are they actually manufacturing it to the specs in a consistent way? It's a little different, and it's a very important difference.

CONSULTANT KRESS: I was assuming you would have a way to when a batch of fuel is made --

MR. RUBIN: Yes.

CONSULTANT KRESS: -- and you're getting ready to load it somewhere, you would take samples of that.

MR. RUBIN: Well, they're doing that in realtime as part of the manufacturing process. And we would just be there as observers to see how they're doing it.

CONSULTANT KRESS: Oh, you could make use of their sampling -

MR. RUBIN: Yes.

CONSULTANT KRESS: -- and testing?

MR. RUBIN: Absolutely.

CONSULTANT KRESS: But doesn't it have to be tested in pile?

MR. RUBIN: Oh, well, the concept of fuel qualification is that you develop and fix a manufacturing methodology.

You say I now have a method that consistently meets my specs and this is how I do it in terms of equipment and procedures and specifications and characterization and QA statistics and so forth, and this is the entire package of how I did it. It's the cookbook.

And now I'm going to test the fuel and in the irradiation accident condition simulation to see if the performance is what it needs to be.

And then if it turns out to be okay, I have qualified my fuel that I made this way, and it will perform under normal operation at the design level of service conditions and at the accident conditions and that's fundamentally my qualification.

Now, I got to start making fuel for the plant that needs to go into operation. And the presumption is that the fabrication facility will then follow that cookbook in exactly the same way and end up with the same quality of fuel and will perform in the same way, but there is no argument today that says I still need to routinely irradiation test the fuel that comes out.

CONSULTANT KRESS: I was assuming -

MR. RUBIN: However, there is one distinction. The qualification that Dave's program is talking about is an industrial scale, but it's not a fuel fabrication facility yet.

I think Dave would say I need to turn this all over to my fabrication plant and they need to have a run of production fuel, and they need to go through it, and then they need to have some small sampling that goes into an irradiation accident condition testing to show that they themselves have had the same result with their fuel fabrication facility as Dave and his people got at the end of his fuel qualification program.

MR. PETTI: And that's in the plan, but it's never talked about in this plan. It's talked about in our acquisition strategy that you need a proof test, but you do not need it routinely.

You would think at that point -

CONSULTANT KRESS: There are some

interesting statistical questions about how can you do this? How many samples do you take? Why would you irradiate them, and where do you irradiate them?

There are some interesting issues there.

MR. PETTI: But by then, you know, we'll be a heck of a lot smarter, you know, by the time we figure out what's the right thing to do.

CHAIRMAN BLEY: Maitri, did you -

MS. BANERJEE: Yes, I did. That's the ORNL documents.

CHAIRMAN BLEY: The ORNL, yes.

CONSULTANT KRESS: Okay. Thanks.

MS. BANERJEE: Thank you.

CHAIRMAN BLEY: Anything else for Stu?

Okay. Stu, thank you very much.

MR. RUBIN: Thank you.

CHAIRMAN BLEY: I guess we're ready for Srini.

MR. BASU: Srini.

CHAIRMAN BLEY: Yes, right? So, we're about a half hour behind.

MR. BASU: Srini is going to help us.

CHAIRMAN BLEY: I'm not so sure. I think carbons and others is the issue, but maybe after that we can begin to catch up.

(Off-record discussion.)

CHAIRMAN BLEY: Okay.

MR. SRINIVASAN: Good afternoon. I am Dr. Makuteswara Srinivasan. I am a senior materials engineer specializing in nuclear graphite in the Office of Nuclear Regulatory Research. I will present an overview of the current status of NRC's research in nuclear graphite.

I will present the technical progress made since my last presentation to you on January 15, 2009. Based on such progress, we have also modified our graphite research plan. I will address all these areas listed here sequentially.

The outcome of our research will inform technical assessment and independently confirm applicant's technical basis for regulatory and safety decisions on graphite core components used in gas-cooled reactor.

Our research will also address data and model uncertainties in graphite behavior in HTGR environment.

We will use the results to confirm material specifications, conformance to codes and standards, and provide information and data for NRC evaluation model.

Our results will also inform staff evaluation of HTGR applicant's probabilistic risk assessment due to graphite component degradation.

Our research contributes to the development of technical staff expertise to conduct safety evaluation of HTGR core component design and ensure the presence of adequate design margin for safe reactor operation.

We have previous licensing experience with Peach Bottom and Fort St. Vrain graphite reactors. Our most recent involvement in graphite research dates back to 2002, when Exelon approached the Commission with a potential design certification application for a pebble-bed modular reactor.

At that time, we developed a document outlining needed research in nuclear graphite to provide technical safety information to adequately review a gas-cooled reactor design.

We terminated our graphite research when Exelon abandoned their plans. We re-started our graphite research in 2006, in response to The Energy Policy Act, enacted by the Congress in 2005.

I have listed the major technical issues in this slide, particularly important for staff technical review.

Good progress has been made by worldwide research in these areas during the past five years. Specifically, the ASTM has published nuclear graphite specifications and the publication of the ASME Code Case for the design of graphite core components in a high-temperature -- and the publication of their Code Case is expected to be published later this calendar year.

Material and component inspection codes are yet to be developed by ASME. When they become available, we will review and then begin an effort to provide guidelines for the staff to evaluate the inspection and evaluation methods provided by the reactor applicant.

Methods are being developed by leading nuclear graphite experts to predict irradiated properties from fundamental material building block. This is a challenging effort.

However, when accomplished, this will hopefully eliminate the need to irradiate and determine properties every time newer graphite is introduced for the reactor core component.

Since the draft research plan was issued in 2006, world-wide research organizations have been addressing most of these issues in their research.

Thus, we expect to use these research results in the development of staff positions and guidance documents.

I have listed here significant items in our graphite research strategy. Our main position is that the applicant is responsible for providing us with full data and technical bases in developing their design of graphite core components ensuring adequate operational safety.

The staff will use these and other data, including the lessons learned from the operational experience, for example, from the Japanese HTTR and the Chinese HTR-10 reactors in safety evaluation.

We will continue to interact with the DOE, Idaho National Laboratory and Oak Ridge National Laboratory research staff as per guidelines established by the NRC-DOE memorandum of understanding, to efficiently and effectively promote and gather technical information needed for safety assessment of graphite components.

We will continue to cooperate with codes and standards organizations providing input on regulatory expectations for addressing uncertainties in the data, and ensuring these are addressed in developing conservative design safety factor.

We will continue to participate in

international meetings, such as the International Nuclear Graphite Specialist Meeting, the NEA/OECD International Cooperative Research Program on Graphite Creep, IAEA activities with respect to irradiated graphite properties knowledge base, and the Generation IV International Forum Graphite Working Group to develop materials handbook.

Such international collaboration has provided the staff valuable exchange of technical information and knowledge on materials and inspection challenges.

Through this international cooperation, we are able to maintain and advance our technical awareness and competency in assessing graphite performance.

Periodically, the staff may identify specific areas where independent confirmatory research may be needed. Subject to resources available, we expect to continue such confirmatory research.

In order to effectively and efficiently review HTGR design, we need to develop the infrastructure and the technical know-how to conduct safety assessment of the applicant's core component design.

Thus, the current challenge for us who

have predominantly light water reactor safety assessment experience, is to develop our expertise to understand and apply material performance codes, component integrity evaluation codes, graphite surveillance requirements and inspection codes, and the tools to evaluate the applicant-proposed graphite component degradation management program.

We do not need to have our own independent codes in all of these areas. However, the staff should have adequate technical expertise to evaluate the applicant's use of codes, and ensure the applicant has provided sufficient factor of safety for designing graphite components commensurate with data and model uncertainties.

Based on the accumulation of such knowledge, the staff intends to embark on efforts to develop regulatory guides which address the evaluation of the structural integrity of graphite components and in-service inspection plans and surveillance techniques which may be proposed by the applicant.

Let me now talk about current status. First, let me address the consensus codes and standards. During 2003, we awarded a contract to Oak Ridge National Laboratory to re-start the ASME graphite code case which had been dormant since 1993.

We also asked Dr. Tim Burchell, the Principal Investigator, to initiate and coordinate new ASTM activities for standardization of nuclear graphite specification and review and update standards for determining graphite properties particularly for nuclear graphite.

Thanks to the dedicated efforts of international participants from industry, including reactor core designers and graphite manufacturers, academia, national laboratories and regulators, we now have the ASME Division 5 Boiler and Pressure Vessel Draft Code Case for the design of graphite core components, and ASTM specification for components subjected to severe irradiation dose, and another ASTM specification for those components which are subjected to limited dose, but nevertheless play a critical role such as supporting the core.

MEMBER REMPE: Excuse me for just a second.

MR. SRINIVASAN: Yes.

MEMBER REMPE: The H-451 supply graphite is no longer available. And so when I hear this discussion, I'm wondering how you're accommodating candidate graphites that may be used to replace that 451, because is there just one set of properties or with the various types of graphite, how are you going to deal with that issue?

MR. SRINIVASAN: Right. You are correct. 451 grade is no longer available. It is not replacement, but what the graphite community did was to issue two ASTM specifications.

And what it is, is that there is a graphite class. This is known as graphite class. if class of material having such and such properties, such and such limits for the impurities and things that all participate in this.

So, any graphite manufacturer that conforms to these ASTM specifications, basically will have - but it's not qualified or something. That's a different thing.

But they have - this is the minimum requirements for room temperature, things that people can do.

Day is gone. Experience gain in the Fort St. Vrain and Peach Bottom reactor experience, as well as the - largely based on AGR experience in the British reactors somewhat, a little bit, but it's GTR and HTR.

Did I answer your question?

MEMBER REMPE: Yes, you did.

MR. SRINIVASAN: Thank you.

MEMBER RAY: Thank you for mentioning Peach Bottom. I forgot to do that.

MR. SRINIVASAN: You're welcome.

MEMBER RAY: I didn't know anything about Peach Bottom personally, but -

MR. SRINIVASAN: Basically, after the Peach Bottom in 1993, there was a court case that was laying dormant with respect to graphite specification itself.

That was a ten-page document in 1993. That's where we were. And here is the latest ASTM one that has gone through the Committee - I mean ASME. Excuse me. ASME design code.

A lot of people put in a lot of effort, and still we are still going through and dotting Is and crossing their Ts.

And I'm sure that based upon the AGC experiments that are going on in INL and other facilities, we will probably revise this. But at least there is a basic document that many people have converged that this is a working document.

But the Peach Bottom and Fort St. Vrain experience are definitely informed writing this document.

All right. Okay. I want to mention that there are also other properties, graphite properties that are important. Therefore, there are also ongoing efforts to devise and update existing property standards and initiation of new efforts in the area of nondestructive evaluation of graphite.

The NRC staff has actively participated in these codes and standards development activities. We have provided input on potential technical safety issues for graphite, and our expectations on defining and establishing the minimum requirements in properties and stress and temperature limits, considering the uncertainties in data, material models, component analysis models and inspection models.

The objective is to better understand the risk contribution for mitigation and proactively establish requirements for compensatory measures if needed.

During 2007, the staff in cooperation with DOE, conducted a phenomena identification and ranking table exercise known as PIRT, and identified several significant safety phenomena pertaining to graphite core performance.

Thankfully, many of the high-importance phenomena with low knowledge are being addressed by DOE and other international research.

These include, for example, the dimensional stability that you asked about, the irradiation creep, thermal conductivity and the Coefficient of Thermal Expansion as a function of irradiation dose and temperature.

Other research being conducted worldwide is expected to provide information on phenomena which were ranked high for safety importance, and for which experts agreed we needed improved knowledge.

We intend to actively provide technical input and suggestions to include potential effects of data and model uncertainties and how these will be addressed in the design of graphite components.

Currently, we have a research contract at Oak Ridge National Laboratory to determine the magnitude of energy stored on previously high-temperature irradiated graphite specimens, and evaluate the characteristics of this energy release during subsequent heating.

From safety perspective under certain conditions, the release of stored energy can start a self-sustaining reaction.

The stored energy is not considered to be a significant safety issue for reactors operating at temperatures greater than approximately 300 degrees

Celsius, which would be the case for NGNP HTGR. This research will provide technical data to confirm this position.

To conduct independent confirmatory analysis of applicant's stress safety margins for graphite core components, we are also developing a finite element stress analysis tool at Argonne National Laboratory.

The code will incorporate ASME design procedures. The code will determine the spatial stresses for reflectors and moderators containing keyways and holes or channels for fuel rods, control rods and coolant flow paths and graphite core supports.

This code will incorporate the inherent nonlinear elastic stress-strain behavior of graphite, spatial variation in temperature and flux, and the contribution and role played by irradiation creep to changes in graphite properties.

The time or dose-integrated stress will use the time or dose-dependent coefficient of thermal expansion, thermal conductivity, nonlinear elastic modulus, Poisson's ratio, and material loss factor due to oxidation or other potential corrosion mechanisms from coolant chemistry changes and dimensional changes as input for stress calculations.

Reactor applicant and DOE research will provide the data. The model and the procedures will be validated and verified using ASME code and DOE and other vendor data and benchmark calculations on idealized core component shapes.

This finite element analysis tool is expected to be available by 2013. We intend to use this tool to confirm applicant's designed deformation limits for graphite core components.

As we move along the NGNP path, we intend to engage DOE research staff and provide technical input as appropriate, by monitoring worldwide nuclear graphite research.

We will complete research at Oak Ridge National Laboratory and Argonne National Laboratory and publish research information in the form of NUREG or other publicly available documents.

We will continue to actively participate in ASME and ASTM codes and standards activities, providing regulatory perspective to information needs and maintaining adequate conservatism in stress and temperature limits for graphite components commensurate with the estimated uncertainty in irradiated properties data and material models.

We will assess the accumulated knowledge with respect to providing adequate information for the structural integrity assessment and inspection requirements for graphite core components.

Based on such information, we will develop and publish regulatory guides incorporating public comment on graphite core structural integrity assessment and guidance on graphite core inspection.

I conclude my presentation.

MEMBER ARMIJO: I just want to make sure I understood.

On, I think, your Slide 6, you said many of the high-importance, low-knowledge things are now being addressed.

Does that mean that some high-importance, low-knowledge things aren't being addressed or was it just the wording?

MR. SRINIVASAN: I think I misspoke.

MEMBER ARMIJO: Everything that you're concerned about in that category -

MR. SRINIVASAN: I'll say a couple of instances where the research is not being done. One of the things that was identified was spalling of graphite.

MEMBER ARMIJO: Spalling?

MR. SRINIVASAN: Spalling. Spalling in the sense if you are thinking of a channel, for example, and there are two cracks that come together in a triangular fashion, let's say, and then graphite falls off or spalls off.

So, in the refractories industries, spalling is a big issue and they do work. Even in the graphite electrode industry they do that.

So, that's a concern that's not being addressed, but the fracture is being addressed in a different format, really. So, that's one aspect.

Other one is with respect to the typology of graphite in terms of the material loss itself. That's not also addressed very well.

Third item perhaps that is important is really that's the post-accident kind of a scenario in the sense that they're comparable to water ingress and air ingress scenarios, really.

The material loss due to air ingress, severe air ingress and water ingress is not being addressed in a formal fashion as yet in the programs, but they are starting to be addressed. Let me put it that way. So, that's my caveat with respect to that.

MEMBER ARMIJO: Okay. Thank you. That's what I wanted to hear.

CHAIRMAN BLEY: I'm just thumbing through your backup slides, and I appreciate you bringing those. We have talked about those issues. I think you covered the stored energy.

Nobody here has pushed on the burning of graphite issue, but I think at the full Committee you need to be prepared to address that. And if anybody here wants to hear what you've got, I looked through it and it's really consistent with things I've seen and - but I think that's something that will come up in the full Committee for sure.

MR. SRINIVASAN: Thank you for -

CHAIRMAN BLEY: I won't push you to go through it now.

MR. SRINIVASAN: I just want to mention, yes, I think just very quickly a couple of slides.

CHAIRMAN BLEY: Okay. Good.

MR. SRINIVASAN: Basically, the burning in place is self-sustaining reaction. For all practical purposes by this definition, graphite does not burn.

MEMBER CORRADINI: So, what occurred at Chernobyl then?

MR. SRINIVASAN: Okay. Good. Thank you.

In the Chernobyl accident, basically it was uranium - sorry. Not uranium. It was cladding that burned.

Material loss is - because in order for the oxidation to occur, you have to have continuous flow of oxygen.

In the real cases, in the HTGR case that we are talking about, the fuel is different and things. In the case of scale, it is the uranium fire basically.

MEMBER CORRADINI: But doesn't that, just so I'm clear, so you're saying that you've got to have another substance that is essentially isothermically adding heat to the system.

MR. SRINIVASAN: Right. And also you're to be replenishing oxygen continuously.

The other thing is that it is also in the press and everywhere, graphite glow and things - the glow of graphite, per se, does not mean burning.

And the so-called A squared on heating business is based on resistance heating. Basically, you supply the current and its resistance with the graphite keeps the graphite.

So, basically on Figure 1 here on the left-hand, far left, we are looking at the graphite coming out of the furnace after being heated to 2800 degrees centigrade. Okay. So, that's being cooled and we see the glow up there.

In the middle one, you are seeing graphite electrode glow in the arc furnace. In the far right-hand corner, basically you have three graphite electrodes.

On the top, you can see, what do you call, greenish color. And then of course the white color and things, really.

The last, you know, so, you can take these things and then they are graphite pristine, so to speak.

So glowing, per se, is really you can think about our gas, you know, what do you call in the home? Washing machine, dryer. No plug is there for ignition thing. That will glow. Glow, per se, does not constitute burning.

Okay. Moving along. Right along. There's also anecdotal things in the Windscale fire, which is called Windscale fire, but it's really the uranium metal fire. And basically they looked at the graphite before and after on things. Really, there's not a -

MEMBER CORRADINI: There were some really extensive reports on that.

MR. SRINIVASAN: Right. So, I wanted to

include that.

And there was also - I'm sorry. You guys didn't ask about dust explosion. So, I'll stop with that.

MEMBER CORRADINI: So since we didn't, let's talk about it.

MR. SRINIVASAN: Oh, sorry. There's also the question about graphite dust explosion. Is it possible, and so forth, really?

Lot of work has been done in the recent past, that is within the last ten years, because of the British. They have decommissioned the reactors and then there's a lot of dust going to be created because the irradiated graphite and what it is and so forth.

So, they did a lot of work in Italy, France, as well as in the UK. And this information is available in an EPRI report. And I think in the last meeting if I remember, I also showed the - I showed at the ACRS meeting a movie of the combustion experiments and so forth.

The bottom line is that it is termed as - it doesn't explode out. It's weakly explosible in the sense you have to have a lot of ignition energy, you have to have closed atmosphere and so froth.

And even then it's "poof" kind of a thing. It is not like a flower or, you know, with green things doing and so forth, that kind of a thing.

The other thing is that again in the Windscale, there are thermal lances used to cut the restraints and so forth. There was no explosion. And dust is invariably produced, for example, in shops like this, typical fabrication shop and so forth where there are opportunities for dust entrapment in isolated areas and things.

MEMBER CORRADINI: But in terms of burning, though, since the length scale is so small, you would get burning, right?

MR. SRINIVASAN: The burning - you would get oxidation. Oxidation. Yes, indeed. Oxidation. If there is oxygen and things, you will get oxidation.

MEMBER CORRADINI: So, the concern about explosion is simply because it's explosive because of the dynamic pressures or the fact because you would get in any sort of a closed environment, you would still get a combustion event of some sort, a burning event?

MR. SRINIVASAN: Burning. And the thing to differentiate is really, for example, in the - I think it was Don Carlson, our colleague, pointed out just after this Japan incident and things, there was somewhere in the press saying that graphite will burn like coal and so forth. That kind of data is a mistaken impression also.

So, if you look at the things, really, the chemical composition of coal is basically will have a fuel component in it in terms of hydrogen and so forth.

For graphite or even graphite dust, it's purely carbon material. So, there is the differentiation that one has to be aware of.

I think I included these things except for these two pictures. One, you probably don't have the glow. This picture is not -

CHAIRMAN BLEY: Yes, that's not in there.

MR. SRINIVASAN: It's not there in your package. I'm sorry about that. Later on - because, again, I saw somewhere that glow is fire and things. So, I thought I better put that in. I apologize for not having included that.

And I also have not included this in the previous one, I guess. Because this is again I wanted you to be aware of graphite, machining is there and so forth. I will be glad to give this to you.

CHAIRMAN BLEY: Good. Anything else?

MR. BASU: That is for the full Committee presentation, would you like us to include this discussion as main part of the presentation or keep them as backup slides and -

CHAIRMAN BLEY: I think the burning one as at least one slide on it in the main presentation, and have the backups in case. Because I know there are some member - the last time we talked about this, there were some members who wouldn't quite buy into what we see here.

So, yes, one slide, and the rest is backup. Yes.

MR. SRINIVASAN: Appreciate that. Thank you.

CHAIRMAN BLEY: Thank you.

MR. BASU: Hopefully, we saved some time.

CHAIRMAN BLEY: Yes, we did. Let's do the last presentation in this package, and then we'll take the break.

MR. BASU: Good. Shah is next.

(Off-record discussion.)

CHAIRMAN BLEY: I know you don't have a name tag, but we were introduced that you'd be here.

MR. BASU: Yes, I will introduce her.

CHAIRMAN BLEY: Okay. Good.

(Off-record discussion.)

MR. MALIK: Good afternoon. My name is Shah Malik. I will be making a presentation on creep and creep-fatigue flaw assessment capability development in high-temperature reactor materials.

After my presentation, there will be a follow-up presentation by Dr. Amy Hull on a related topic of monitoring for management of materials degradation that concerns flaw monitoring, flaw detection in high-temperature reactors.

The objective of this work is to develop an independent confirmatory capability for creep and creep-fatigue crack growth elimination process in high-temperature reactor material that are operating when the creep strain becomes significant at high-temperature range.

This will assist NRC in developing regulatory technical basis, as well as assessment of licensee submittals on integrity of structural component operating in creep range.

Just I would like to remind that creep is not a deformation that is operating for LWR components, because their temperature is much lower.

Breach to secondary system due to creep or creep factor crack growth may occur in intermediate heat-exchanger/steam generator, cross vessel. Or if the temperature are high enough in reactor pressure vessel, that could lead to develop pathway for fission product release, as well as air, steam and water ingress.

NRC PIRT, the NUREG that was conducted in 2007, 2008, NUREG/CR-6944, determined a creep factor flaw growth phenomena to be of high importance and low knowledge.

In particular, creep and creep fatigue can occur in structural discontinuities, as well as in weldment operating in creep range of temperature.

The flaws can be preexisting or they initiate early in life. If those flaws are not detected or repaired after, they can grow sub-critically under creep fatigue deformation.

And they can grow to critical size that can trigger a structural failure mode as shown by the middle bubble there, leading to component failure and breach of primary or secondary system, and finally a fission product release.

Current creep fatigue crack growth modeling which is currently available, we can see there's secondary creep.

On the left-hand side, I have shown a

schematic for creep strain versus time for a given temperature and given the stress.

Initially, you have a primary transient creep followed by a steady state secondary creep regime. And, eventually, a tertiary creep when the creep rate become much more higher leading to creep rupture.

Current classical creep or fatigue crack growth model only consider the secondary part of the - which is the steady state part of the creep curve.

The second graph shows there an actual data for chrome-moly vanadium steel at 550 degrees. As you can see, there is a significant secondary creep that is straight line creep slope followed by quite a significant tertiary creep also.

Now, there is a British R5 methodology, French RCC-MR, as well as American Petroleum Institute methodology for this creep crack growth under a steady state secondary creep.

And that is applicable to chrome-moly vanadium, as well as the stainless steel 3/4th, 3/16 steel. Applicable to fossil power plant, as well as previous UK AGR programs.

So, the current time-independent creep procedure will not be applicable to high-temperature metallic material that exhibit time-dependent creep response. So, secondary creep is very - is non- portion of the creep curve.

Here you have shown two different material; Alloy 800H, which is one of the candidate material for current NGNP project, as well as Alloy 617 for a little bit higher temperature range. And both of them do not show -- for example, the Alloy 800H shows creep strain rate. We do not see a significant part of the secondary or steady state uniform creep rate.

Similarly for the Alloy 617, there is a small portion of secondary creep followed by a significant tertiary creep region. This was developed - Alloy 617 was developed under the NGNP program. so, it's the current data. Whereas Ally 800H is some earlier data.

So, this is for us to develop - there's a need to develop a time-independent creep fatigue crack growth modeling procedure that can be applied to creep deformation when there is significant time-dependent response of creep.

Time-dependent creep in nickel-based alloys such as 800H and 617, and some higher- temperature alloys such as Haynes 230 and additional alloy hastelloy X and XR, have a schematic similar kind of behavior.

They start out as a creep strain rate with this time plot. You have primary creep. It is decreasing with time. Creep rate is decreasing with time.

Then there is a creep softening region where it is slowly increasing creep rate. Followed by tertiary creep.

Now, this is assuming there is a crack present in front of the material. Then after application of load, initially there will be elastic stress field.

As the time is progressing, there will be a primary creep zone head of the crack tier forming. And within that creep zone as time increase, a secondary creep zone will form. And with increase in time, you can have tertiary creep.

We have tried to come up with a roadmap for developing high-temperature time-dependent creep-fatigue flaw evaluation computer code. There are three different modules in this computer code, hopefully.

Methods of Development module which is shown on the left-hand side, it has three components.

The first basic component is time-dependent fracture mechanics methodology development in which we are developing crack tip parameters and crack tip constraint effects for creep when all the three components of creep are present. There is primary, secondary, as well as tertiary creep.

And that will be applied to develop crack growth correlation for proof of concept, which is Step B of the same module.

And then apply it to more representative geometry such as slender with cracks to develop crack growth correlations.

And develop implementation strategy, the final step in the method development module, so that extrapolating this to longer service time.

For NGNP-specific application, there will be a crack growth data development phase. That will develop crack growth data.

Now, I'd like to mention very specifically the data originated essentially with DOE and applicant. We are using those data in our application to develop correlations.

So, the left upper side, top portion, Section D, we are using those data to develop correlations in the air, as well as in pure helium.

Finally, implementation module. First, we'll develop a deterministic flaw evaluation procedure. And using data uncertainty and other probabilistic information for the material, as well as modeling and incorporate it into an NRC modular probabilistic fracture mechanics code.

I think you may have heard about currently it's called XLPR code, extremely low probability of rupture code, that is being used for dissimilar metal weld evaluation in primary cracking in PWR components.

So, there will be further development on that in the next three, four years. And that's with this code we are going.

Now, the main work, which is the time-dependent fracture mechanics methodology development, that is being done at Oak Ridge under Program N6654.

It uses what is called concept of similitude between lab specimen and the actual stress component.

So, the equal stress and deformation states in lab specimen will imply there will be growth stress and deformation state and crack tip of actual component.

That will then transfer crack growth data from specimen to structural application subject to applying appropriate crack tip constraint when we're going from specimens to a structural component.

Data generation, as I briefly mentioned before, will be the responsibility of DOE, as well as applicant.

There are three sets of data that go into - initially it will be scoping data. And then finally design data. When they are assembled completeness, they will have both the creep crack growth data - when I say "creep," I mean both creep and fatigue crack growth data. And there is mechanical creep strain data in air to develop time-dependent fly release methodology.

And to determine an effective environment such as impure helium, as well as effective steam. There will be scoping data again developed at INL, DOE, as well as at applicant.

And to find the affect of thermal aging on crack growth initial scoping data, and then final data set combined together for multiple heats to develop design data by the applicant, as well as DOE labs for both base metal, as well as weldments.

MEMBER ARMIJO: Is that data being developed for, let's say, the 800H, the -

MR. MALIK: The data is being developed for

800H, but INL is also developing for 617 as well.

MEMBER ARMIJO: And the other, the Haynes 230 and the Hastelloy?

MR. MALIK: They are not being considered at the moment.

MEMBER ARMIJO: Okay.

MR. MALIK: But they are the same class of material if they were to - applicant were to use for any component, they will end up developing some on top of that as well.

This slide shows the major milestones. We are working on the top row. Time-dependent fracture methodology development. Soon, this fall, I think we'll have the methodology completed.

We are also performing the peer review independently by external reviewer to peer review to use it finally into the next step of the process.

The second two rows shows developing crack growth correlation starting in last quarter of '11, going to '12, as we go up to fiscal '13 year.

And then in parallel, there will be NGNP-specific crack growth generation at INL, as well as the Office of Contract at Argonne National Lab for creep crack data generation between '12 and '14.

And hopefully implementation of crack

growth as the data start becoming available between quarter '12 to quarter '14.

Now, here we have - I'm going to pass it on to my colleague, Dr. Amy Hull.

(Off-record discussion.)

MS. HULL: Good afternoon. In January 2009, we presented you some of the highlights from the 2007 PIRT analysis in elevated temperature materials, and identified improved in-service inspection to monitor for management of materials degradation as being of high importance and low knowledge.

Traditional ISI methods used for LWRs may be insufficient to monitor development of cracks and materials degradation in the NGNPs because significantly longer periods between outages and the potential for high-operating temperature-induced materials degradation may require more than can be provided by - and those techniques we'll talk about.

We've been looking at these issues since that time, but the activity has been resource limited. Consequently, I will make this presentation brief. There are only three short slides.

The impetus is derived from work within the RES Division of Engineering dating from the NDE Workshop for events reactors held in September 2006, hosted by INL.

The NRC staff that participated came away with the message that some of the key issues were ASME code and new NDE inspection concepts, applications and challenges.

It was at that time, also, that we became more aware of the RIM, the Reliability and Integrity Management approach, that was being considered for rules for inspection and testing of components of gas-cooled plants.

As a result of this, since late 2006 I have actively participated in ASME boiler and pressure vessel subcommittees and working groups that have focused on events reactors.

The Gen-4 NGNP materials project had a number of tasks. One of the tasks, 12, in-service inspection technology for HTGRs, resulted in WCAP-17084-NP. That further highlighted the value of online monitoring such as acoustic emission.

Since then, we've continued our assessment of AE monitoring of structural integrity looking at fatigue, stress corrosion cracking, creep and leak detection for high-temperature gas HTGR materials.

PNNL has experience with crack monitoring in a reactor environment. So, because we were working with them on activities related to the proactive management of materials degradation known as PMMD, and we published a paper collectively at ICON 2009 Analysis of Emerging NDE Techniques, Methods for Evaluating and Implementing Continuous Online Monitoring, we went to PNNL for a new contract.

A sister report, PNNL 19401, Methodology For The Analysis Of Emerging NDE Techniques, provides more detail. As well, it discusses the AE validation on an operating reactor. PNNL was involved with monitoring SEC in the Limerick Unit 1 BWR Mark 2.

Next slide. For the 22-month project, Effectiveness and Reliability of Acoustic Emission For NDE in Advanced Reactors, PNNL personnel work with the ASME code committees to update AE standards for online use in either leak detection in DE technology or flaw detection in monitoring.

Working with the NRC group, they review the standards currently being generated by the Section 11 SWG HTGR, the special working group for HTGRs.

Together, we're identifying potential issues with these standards and identifying where additional information may be needed.

Based on extensive previous experience, the PNNL group is determining the type and extent of evaluations working with us that we would need in order to review licensee-proposed ISI programs.

And then finally as part of this work N6907, within 22 months they'll provide us a basis document for a draft regulatory guide for the ASME Section 11 inspection programs for HTGRs.

Task 1, reviewing the ASME proposal for RIM, is important. This group was led by members of PBMR such as John Fletcher and Neil Broom, and Technology Insights such as Carl Fleming, Steve Gosland and Ron Campbell.

In developing the RIM methodology, they started with some of our reg guides such as 1.178, an approach for plant-specific risk-informed decision making ISI of piping, but they're working with the May '98 draft.

When I joined this committee in 2007, they had a rough draft done of this. Now, it's at the point where it can be reviewed by the special working group. And we are reviewing it and making recommendations first.

The second task listed there is to analyze the existing ASME code for application to advanced reactors and make recommendations regarding the use or modification of existing AE provisions for use as leak detection, crack detection and crack monitoring techniques.

The use of online or continuous monitoring methods can be an effective way to ensure public safety and component integrity.

AE is a prime example of a good option in this context. It can monitor the region of piping or pressure vessel. It can passively listen for ultrasonic signals of a hundred to 400 kilohertz emitted by cracks as they grow. And further, is sensitive to cracking, but not to corrosion.

Next. As mentioned in the previous slide, we are participating in the special working group developing the Div II rules for inspection and testing of components of gas-cooled plants, which complements the Section 11 Div I parallel rules for inspection and testing of components of light water-cooled plants.

The Div III as it's developed, will be for liquid metal-cooled plants. And I'm NRC's rep for the Div II and the Div III.

In the 2010 code that was recently published, the wording for Div II still takes only one and a quarter pages. The forward consists of historical background and in-service inspection philosophy and another subcaption, reactor system considered in co-drills.

So, in other words, developing Div II was absolutely wide open for whoever was involved on the consensual committees.

And in this case, as I mentioned before, the people who were invested in this were the representatives from PBMR and Technology Insights.

So, when they developed the IGA, this is the draft IGA, general requirements for gas-cooled NPPs, there was a focus on sort of a risk-informed approach.

This 118-page IGA general requirements has been drafted and is out for pre-special working group review. It's not out yet for balloting or pre-balloting review.

The activity has slowed down with the loss of the former PBMR participants. So, we have more time to revise the document as needed.

Participants of the ASME special working group propose that the RIM program inspection roles and requirements of this division will apply to all passive metallic pressure-retaining components and their supports, including pressure vessels, piping, pump casing, casings for turbines and compressors, supports and valves.

Acoustic emission is specifically encouraged in the proposed IGA general requirements, the IGA-2934 acoustic emission monitoring and examination.

The use of RIM, reliability and integrity management, and the general requirements for the rules for inspection and testing of components of gas-cooled plants, seems to be a new approach.

The first sentence of IWA-110 scope for light water-cooled plants states - this is the Division 1 - this division provides requirements for in-service inspection and testing of light water-cooled nuclear power plants.

On the other hand, the proposed IGA-1100 scope begins, this division provides the rules and requirements for the creation of the reliability and integrity management, RIM, to select and implement a combination of design, fabrication, inspection, surveillance, operation and maintenance requirements that meets the plant-level risk and reliability goals that are selected for the RIM process.

So, my short overview is to sort of indicate that I think a careful review by our NDE and ASME code specialist is well-warranted since this seems to be a significant change.

We in DE, are also consulting with PRA specialists. The team at PNNL working on this include subject matter experts who have focused for a number of years at the interface of NDE materials degradation and PRA. And that's what I have.

CHAIRMAN BLEY: Thank you.

Do you have more?

MR. MALIK: One last slide.

CHAIRMAN BLEY: Sure.

MR. MALIK: Going forward, we will continue with our research effort as per our planned roadmap and interact with DOE/INL and other labs through the NGNP project, as well as participate in codes and standards activity.

CHAIRMAN BLEY: Thank you.

Anything from the committee?

Thanks very much for your presentations. I think we'll take our break now. We will start promptly at 3:30. So, get back a minute or so early, and we'll go ahead.

Thanks very much. We're recessing for 15 minutes.

(Whereupon, the proceedings went off the record at 3:15 p.m. and went back on the record at 3:30 p.m.)

CHAIRMAN BLEY: Excuse me. We're about to start. The meeting will come back to order. Jose Pires will take us through the structural discussion - oh, and seismic.

MR. PIRES: Yes, structural and seismic.

CHAIRMAN BLEY: Excellent.

MR. PIRES: Good afternoon. I'm Jose Pires from the Structural Engineering Branch. I am a senior structural engineer.

This presentation was prepared by several people. We have been involved as program managers for some of these projects. We will be talking about structural and seismic research.

The objective of this research is to develop data and information, as well as to ensure analytical capability for review and for independent confirmatory assessments of safety-related structures and equipment of future nuclear power plant designs.

The scope of the presentation, what we will be addressing, is mostly the effect of high temperature and radiation on concrete, especially high temperature on concrete as a material, and as a structural component.

And then we will be discussing various seismic response issues related to structures and the calculation of loads on components.

And we'll also be talking about the reactor vessel, especially core graphite components and core support structures, and their response to seismic loads, dynamic loads.

On the first topic of concrete subjected to high-temperature demands, we have completed one part of the research. We did a review of effects of high temperature on concrete as a material during heat and cooling. This report is being completed and has been published.

One part of the study was to review what are provisions on current US codes and standards, as well as international codes and standards like what was used to call the European Concrete Code, and now is called the core published by the International Federation of Concrete.

We have a project starting on soil-structure interaction of deeply embedded or buried structures. That project is going to continue.

Other projects that have already been done, this is full competition procurement. The proposals are currently being reviewed for awarding a contract.

We also have another contract with a

national laboratory for multi-module response. That is a condition in which you will have a common foundation and various modules on that foundation, so that you can - the number of modules can vary to the life of the structure that look at these aspects.

What we have not yet started is the evaluate assumptions and assess limitations of existing codes for how can they be used to assess the seismic response of reactor vessel graphite components and core support structures.

These research interacts with the previous work that was presented by - before that relates to the evaluation of mechanical properties of the graphite elements.

Now, this slide here is just to illustrate that for some of the designs it is anticipated that at least the reactor building and other - sometimes even steam generator or other components would be underground. Would be fully buried.

So, for that reason we needed to investigate, and I put on the next slide, we need to investigate what is the current provisions that exist in the current regulatory guidance that are sufficient to address issues that may arise for the seismic response of deeply buried or embedded structures.

The one concern here is to calculate what are the ground motions of some of those components on the vibrations of the ground are amplified because of the structure and what loads they are going to compose on the components. So, we need to assess that for when the structures are deeply buried or fully embedded.

We'll also be looking at the expressions on the walls of these structures. Because as they move and they push against the soil, the soil resists this movement and induces forces, large forces on the walls of the embedded structure.

Other aspects, for instance, if you scroll down below, you'll see the multiple modules on common foundations. And the number of modules may vary during the life of the facility.

This of course is going to change the natural frequency of the combined structure. May affect the interaction, may affect the load on the treatment, what we call the in-structure response factor.

So, we want to look at the provisions that currently exist, how adequate they are in terms of broadening this. We have provisions to broaden the flow response spectra, how this effusion to the - how can these things be accounted for?

Going back here to the concrete and the high temperature, so that report that we just published looked at the physical property assessment methods and -

MEMBER RAY: Wait a second.

Could you go back to the slide where you-

MR. PIRES: Sure.

MEMBER RAY: Either I wasn't following you word for word - on the last bullet, nonlinear dynamic response of graphite reactor internals.

MR. PIRES: Yes. What the - we have not started that. What the issue is there is that some of these internals, as we understand, they have graphite blocks that are attached to each of the dowels, simply with dowels and with some keys to the ones on the side vertically with dowels.

So, this is not a continuous structure. So, this is just an example. The others will be a part of pebble-beds in which you can have densification off of the bed during a seismic event.

MEMBER RAY: Right.

MR. PIRES: And this blocks the -

MEMBER RAY: I understand, but I'm just questioning that that's the only issue related to these structures. In other words, their nonlinear dynamic response.

How about their behavior over time, the ins-service inspection, basically? How do you verify what their characteristics are over time?

Is that an issue?

MR. PIRES: It is an issue, and it is an issue that is being also investigated again by the other project that was referred. It looks at mechanical properties of the materials, but we have not been looking at that aspect in this project.

MEMBER RAY: It's being investigated by the other project?

MR. PIRES: I think that if I understood well on the previous presentations, I saw that they had programs on assessing what are the mechanical properties of the graphite components and -

MEMBER RAY: Yes, that's pretty traditional at least in the 30 years I've been involved in it, but there's always a problem of how do you determine what the condition of those structures are over time? And that's what I'm asking about.

MR. PIRES: Yes, it is I - we have not looked at that, at that aspect yet.

MEMBER RAY: Well, you know, I can't do

anything more than to say I think it should be an issue for somebody, unless somebody's got a response or an answer to it, because I've never -

MR. BASU: Harold, if I can just make an attempt to answer your question, the materials program and the graphite program that you heard in the last couple of presentations, they will be looking at the material degradation from the environmental factors, as well as the aging issues that you are bringing up.

MEMBER RAY: Well, I did hear them say they were looking at corrosion, but I guess I'm - it's fine. If that's going to be the place where the concern I have is addressed, that's sufficient, but it's a little different than just either corrosion or aging. It's actually verifying the integrity of a critical structure periodically during its life.

And all I heard them talk about in that regard is the pressure boundary. And so, that's why I bring it up here thinking it was going to be talked about in this context, but it's not.

It doesn't matter to me, but the point is there's always been as long as I've been involved, an unresolved issue which is do these things ever need to be inspected? If so, how are you going to do it? To what criteria?

Or are you just going to assume you know how they're going to behave over the life of the plant, and that's the end of it? And that's what I'm asking about.

MR. BASU: And, you know, we do appreciate your question. There is the issue of in-service inspection.

MEMBER RAY: Right.

MR. BASU: And you only got a very brief presentation on that through the reliability and integrated management program that Amy talked about in the context of materials.

MEMBER RAY: Well, just make sure it's on your list explicitly, okay?

MR. BASU: Good.

MEMBER RAY: Because it's been a dilemma for a long time. And if somebody has solved it, I would be pleased with that, but I have never heard that anybody has.

MR. BASU: I made a note of that and will-

MEMBER ARMIJO: I'd just like to follow up on something Dr. Pires mentioned, that is the response of, let's say, a pebble-bed core to a seismic event as far as densification of the core, some reactivity insertion.

Has that already been addressed or is it being addressed in another part of your research?

MR. PIRES: We have not started that specific study in this research. It's one of the plans of the research is to address that. We have not started.

There are computer codes that have modeled similar phenomena on densification of granular materials. So, those exist, but -- and I think that there have been studies done by some of the - either national laboratories or some of the developers of these technologies to look at these aspects as well, but we have not ourselves started looking at that yet.

MEMBER ARMIJO: It seemed like that experimental technique that we were shown earlier using these lasers to look at a two-dimensional slice of a three-dimensional structure combined with some seismic shaking would tell you -

MR. PIRES: Yes.

MEMBER ARMIJO: Could quantify a code and say, yes, these things can densify and they will change the -

MR. PIRES: I think that those techniques can be useful to help validate codes, yes, those techniques.

MEMBER ARMIJO: But that's not necessarily being addressed right now?

MR. PIRES: Not yet because we have not started these.

MR. PETTI: We have done some.

MEMBER ARMIJO: You have? Okay.

MR. PETTI: And it's in the published literature.

MR. BASU: Sam, those are addressing the structural issues, but there are obviously other issues, other facets of -

MEMBER ARMIJO: Yes. If you just shake a bunch of balls together, they're going to compact. They eventually get to -

MR. PIRES: Those techniques will be - when I saw the presentation, I thought that those techniques would be useful -

MEMBER ARMIJO: Yes.

MR. PIRES: -- to help validate the codes determining actually initial conditions and to help validations and verification. Probably also inspection, depending how -

MEMBER ARMIJO: Yes, not too expensive. Okay.

MR. PIRES: So, in this area here we have published a review, a NUREG report under the contract to Oak Ridge National Laboratory on the effects of high-temperature on properties of concrete.

Typically, we have been looking at the traditional properties of the concrete that tell the compressive strength, elasticity, modules of elasticity, bond strength.

And part of the work also will involve lessons learned from concrete facilities subjected to high temperatures.

And we are continuing the work to understand how sufficient are the current code provisions and guidance and what modifications those are needed for those for review. So, it's still necessary to continue the activity.

This is just a picture that gives you an idea. It is not present data, but the compressive strength of the concrete is very stable at to about 400 degrees Fahrenheit, about 200 degrees Centigrade.

Same thing for the modules of elasticity, but it does degrade quickly after that. So, you can at about 700, 800 degrees Centigrade get to about 20 percent or 40 percent of the values.

This is just a picture just to illustrate that a phenomena - this is a phenomena in which traditional concrete - I mean concrete with an ordinary cement paste was given to very high temperatures and it dehydrates, and then it re-hydrates again. It's spall, as shown on the right picture.

So, it's just to see that other phenomenology may also be important. Because if you spall the concrete, you then have easier access of the moisture and the temperature to the reinforcement.

We have been looking at various phenomena, but these also can be prepared by using different cement pastes.

The seismic soil-structure interaction issues, as I said, we have several proposals in the review for a project on deeply embedded structures.

The goal is to collect and summarize experimental data. It will build and continuing on previous work that has been reported on two NUREGs.

The first one there, the 6957, already in a comparison in a benchmarking against some experimental data from Japan from JNES with a small scale. A 1/10th scale.

We will be doing a parametric study. The parametric study is going to be looking at things like the ground motion intensity including beyond design basis, advantage and disadvantage of linear versus nonlinear analogies for review, what are the limits of publication of each one, effects like gaps that may develop between the soil and the structure, friction at the interface between the soil and the walls and some other three-dimensional impacts.

Some of the data in Japan is for rectangular cross-sections. And while here, some of the buildings may be more of cylindrical shape. So, those also have some effects on the BA here.

In the entire research that we have been doing, there are some assumptions that some data will be available from the current DOE research work.

For instance, in the case of the elevated temperatures, not only you are looking at the effect of the temperature on the concrete as a material, but we also need to start looking at how the temperature from the reactor - or what will be the temperature on the actual walls and what will be the cyclic soil flow in high temperature, because these have an effect on the behavior to see how much through the concrete wall the temperature will go. So, we need some information on that.

We have been looking at international codes. As I mentioned, European codes for concrete, as well as the American Concrete Institute codes, ASME codes.

We already have programs to obtain data from the 2007 earthquake known as the Kashiwazaki-Kariwa earthquake. And most likely we will try to analyze data from more recent earthquake.

And I did make this summary of what we did. So, we have completed the first phase of the project on the high temperatures. We did a review. We also made a comparison on those reports among provisions and current codes.

We are now analyzing - have a project on deeply embedded structures that builds on two previous projects that is expected to start at the end of this fiscal year.

We have already a project ongoing with multiple models on a common foundation that started already with a national laboratory.

The aspect of the graphite components and reactor support structure, that is planned for a future activity.

We have also worked with the code committees; ACI 349 for concrete structures that are safety related, the AISC N690 which we are developing guidelines for steel plate-reinforced concrete structures, ASCE 43-05 which has information on performance-based design or what constitutes an adequate seismic design basis load. So, we continuing working with the standard developing organizations and committees.

Going forward, so we will continue with the R&D plan that we currently have. We will adapt the needs of that plan based on technology selection if those - because adaptations are needed, you know, particularly for some structural aspects may require more knowledge of what the actual architectural engineering aspect of these facilities would be.

And we will continue to brief the ACRS periodically. And we will of course coordinate with DOE and INL on what are the gaps that we will identify for the reviews. And that pretty much concludes what I have to say.

CHAIRMAN BLEY: Can I just take you back?

MR. PIRES: Sure.

CHAIRMAN BLEY: This isn't my field at all, but to your Slides 8 and 9. I've been staring at them and trying to make sense of the two.

MR. PIRES: Okay.

CHAIRMAN BLEY: I guess we struck a line through these two sets of data that isn't quite the story. So, if you draw a vertical line at any temperature looking at the one hour, four hour, 24 hours and seeing how far these things stretch apart is more related to what we see on the next page that -

MR. PIRES: The next page is different phenomena, but you are right. That line is not just - there are other effects there like the time of heating and - but - and this is supposed to - are the properties at that temperature when it has been heated.

The next page is a little bit different. It's another phenomena that is, well, at these very high temperatures, the concrete when heated with ordinary cement paste, dehydrates, but also there are chemical changes on the -

CHAIRMAN BLEY: That continue after it dehydrates.

MR. PIRES: Yes, they continue after, but then it hydrates. With additional water some of the molecules expand, and that difference in what expands and what does not expand causes the cracking.

CHAIRMAN BLEY: And this is just pure concrete specimen.

MR. PIRES: It's just concrete. Is not reinforced. Is cement and aggregate. Cement and aggregate. And these deformations mostly happen within the cement and the bonds between the cement and the aggregate.

But the other test we have done where if you put, for instance, slag as part of the cement paste, initially then this effect minimizes or almost disappears. So, there are additives.

But the review, we just put that there to illustrate that is not just the traditional properties of strength and - of stiffness, but also these effects that we need to be looking at.

Of course this is very high temperature. So, maybe we will not get to those temperatures in the concrete itself.

That's why I said that part of our knowledge will be to try to look at how all the heat from the reactor actually couples to the walls. And then we will know if a more realistic assessment of what are the temperature ranges we need to -

CHAIRMAN BLEY: Okay. Well, they are different. But if you look at one of those curves, like the first one with the compressive strength, and you just pick a temperature like somewhere around 500 and look at the one hour, four hour and 24 hours, you see it's losing its strength over that time period.

MR. PIRES: Losing its strength.

CHAIRMAN BLEY: Yes.

MR. PIRES: Part of the work is how long you stay in the sustained temperature and also the cycles, how you go up and down.

CHAIRMAN BLEY: And this doesn't show you the timer temperature. This is time after you were at temperature.

MR. PIRES: I think this is the time at temperature.

CHAIRMAN BLEY: I'm at temperature after you guys got to that temperature.

MR. PIRES: Yes, this is at temperature.

CHAIRMAN BLEY: Okay.

MR. PIRES: And this data on these figures is already old. There are more recent data points.

CHAIRMAN BLEY: So, this is just telling you that you need to have concern. It's not telling you what to do about it yet.

MEMBER ARMIJO: To stay at a low temperature.

CHAIRMAN BLEY: Yes.

(Laughter.)

MR. PIRES: Preferably try to stay below 200 degrees Centigrade.

(Off-record discussion.)

MR. BASU: Do you want me to repeat?

CHAIRMAN BLEY: If you want to.

MR. BASU: Well, I said as long as you stay below 200 degree, you should be okay. So, that's the message that -

MR. PIRES: There are specific provisions in the codes at other temperatures that other properties - tensile strength, you know, compressive strength, all those vary in a different manner, but just to give you a range of what happens.

CHAIRMAN BLEY: Okay. Thanks.

Anything else?

Thanks very much.

MR. PIRES: You are welcome.

MR. BASU: Okay. The next one is the PRA on the list.

CHAIRMAN BLEY: Wonderful.

(Off-record discussion.)

MR. WOOD: I'm Jeffrey Wood. I'm going to be presenting on the HTGR research plan for PRA. Mary Drouin is also a support in this work.

MS. DROUIN: I'm here so you all don't beat him up.

MR. WOOD: Thank you, Mary.

Two PRA activities I'm going to mention today. The first is a draft reg guide that was completed by Research and transmitted to NRO for internal review.

This draft guide discusses a risk-informed approach to identifying licensing basis events. And this draft guide is being considered with other ongoing activities in developing a new risk-informed regulatory structure for advance reactors.

Now, those activities are being led by NRO and there's no more work in this research plan in that area. So, I'm not going to discuss it further today.

CHAIRMAN BLEY: Okay. But that's what we would have heard from when we talk to NRO about the small modular reactors and the non-LWR small - SMRs, is this plan.

MR. WOOD: Yes. Right.

CHAIRMAN BLEY: And that's actually under development?

MR. WOOD: There's ongoing activities, yes.

CHAIRMAN BLEY: Okay. Go ahead.

MS. DROUIN: But the reg guide was to support non-LWRs, not specifically SMRs. It was to support the NGNP licensing approach. And a draft reg guide was sent over to NRO for their review.

So, right now it's in their hands until they come back to us with comments, you know. We don't know what will happen.

MS. BANERJEE: I guess what you may be thinking is the White Paper, LBE White Paper that NGNP developed that was provided.

CHAIRMAN BLEY: It's also the appendix to the reg guide. It's the same thing.

Go ahead.

MR. WOOD: Okay. Thank you.

The activity I'm going to be focusing on is a planning study that was undertaken to identify potential PRA research needs to support the HTGR licensing.

And I'll go through the background approach, observations and insights and give the current status of this study.

So, the background for this planning study was motivated by the notion that risk insights will be a part of licensing the NGNP plant. This is pointed out in the NGNP licensing strategy report.

So, the purpose of our study is to identify where there are gaps in the guidance methods or data that's needed to support the development of a PRA model for an HTGR.

The scope of the study is to address all hazards, all operating states, all risk levels and all plant design life cycle phases.

CHAIRMAN BLEY: So, you're focused on what would the PRA look like. And that first bullet up there about how the PRA will get used in the licensing isn't part of what you're reporting on.

MR. WOOD: That's correct.

CHAIRMAN BLEY: Okay.

MR. WOOD: And our concern is, is the PRA technically acceptable?

CHAIRMAN BLEY: Okay.

MS. DROUIN: But how it will be used is an important aspect you'll see later on, because how it's used will depend what gaps you have.

CHAIRMAN BLEY: Absolutely.

MR. WOOD: Of course we have to consider the application that the PRA is being used for.

CHAIRMAN BLEY: Okay.

MR. WOOD: So, the approach to our planning study was to review available literature, to identify unique design and safety issues associated with HTGR that may impact the PRA model.

Also, to identify any gaps in guidance data, methods, models, tools, that are needed to establish the PRA technical acceptability. And to identify the assumptions that will be needed in order to develop a PRA model.

It's particularly important when you're talking about a plant that's in the design phase and has limited applicable operating experience.

CHAIRMAN BLEY: We spent the day listening to a lot of places there's technical gaps that are being filled.

How integrated is what you've done in looking at the PRA, how integrated is that with all these other areas that people are actively working to fill in the technical gaps?

MR. WOOD: Well, we keep in regular contact with the other groups. But the planning study that I'm talking about here is really a comprehensive listing of the things that can affect the PRA.

CHAIRMAN BLEY: Okay.

MR. WOOD: Some of them do already have ongoing research programs and other technical disciplines that will influence what we need to do to have the knowledge we need to review a PRA.

CHAIRMAN BLEY: All right.

MS. DROUIN: Those gaps we don't get into because we already know that there's an ongoing program.

I mean, you know, there's a lot of deterministic stuff that supports the PRA. So, where those deterministic analyses are being done that would support us, we don't get into that because we already know they're being done in another program.

So, it's not a -

CHAIRMAN BLEY: You don't know how they're going to finish though. You don't know if they'll be resolved.

MR. WOOD: That's why they're on our list.

MS. DROUIN: That's why there's close contact.

CHAIRMAN BLEY: Okay. But they are on your list.

MR. WOOD: That's right.

CHAIRMAN BLEY: Okay. Go ahead.

MEMBER RAY: Well, let me say this: This morning it was pointed out that the research program was geared to the EPA description with respect to, for example, siting. And, therefore, it assumed siting at a DOE facility and so on.

Is that relevant to the scope of what you're talking about here?

In other words, has the site location got any - are you looking for any site or are you just talking about a PRA as it would apply in a particular location such as Idaho?

MR. WOOD: Well, the site wasn't a major driver in our review because most of the elements of the Level 3 PRA or the consequence analysis, are pretty much technology neutral. They don't really depend on it being an HTGR or LWR.

MEMBER RAY: Well, what I'm thinking about is how you would approach a PRA for a site that had a very close-in footprint, for example. A small exclusion area in an industrial location.

MS. DROUIN: Well, we -

MEMBER RAY: And I'm just wondering if that would have any bearing on what you're thinking about now.

MS. DROUIN: It depends at what part of the process in the PRA you're looking at for the gap, you know.

In the design stage where you don't have a site, then the assumption is made for a generic site.

As you move more into the operational, then, yes, where that site is located becomes important.

MEMBER RAY: Well, yes, whether it's in a remote location or in a -

MS. DROUIN: Exactly.

MEMBER RAY: -- area in which you can't assume a large exclusion distance and that sort of thing.

Okay. Well, fine. If it doesn't have any bearing on what you're doing right now, that's enough of an answer.

MS. DROUIN: Well, it's one of these - it will be what we call a high-level finding, and Jeff will be getting into that.

MEMBER RAY: Okay.

MR. WOOD: Okay. So, let me just quickly go over some of the documents that we reviewed for our planning study.

We looked at the technical documents on the - discussing the HTGR design features, some NGNP studies and also past information submitted to NRC for previous HTGR design such as the MHTGR and the Fort St. Vrain reactor.

We looked at standards documents related to PRA. We reviewed the published ASME/ANS PRA standard for operating reactors covering Level 1 PRA and LERF.

We also looked at several PRA standards that are under development covering different aspects of PRA for both LWRs and non-LWRs.

So, we have two types of observations coming out of this study. We grouped them into high-level issues dealing with the overall PRA technical acceptability, and detailed issues related to each of the technical elements that make up a PRA.

So, these are just examples here on this slide. We have more examples documented in a report for our planning study.

Examples of high-level issues are observations related to the acceptable level of detail for PRA, the risk metrics and the associated criteria that are used to assess the results of PRA.

The spectrum of the operating states that are analyzed, and also the hazards that are analyzed, and any unique impacts of those hazards.

The example is given here of a seismic event impacting the prismatic blocks that could clog cooling channels. We heard more about that in the last discussion.

So, some examples of the detailed issues. And these are potential issues that may need additional research to address.

Examples include things like the analysis of potential initiating events and derivation of initiating event frequencies, defining PRA success criteria. Also, considering that the PRA is intended to be expanded to include sequences for anticipated operational occurrences, design basis events and beyond design basis events.

Understanding of unique environments and conditions for an HTGR that can impact the system performance.

There are human performance considerations such as controlling multiple units from a single control room, and modeling of new phenomena in the PRA. For example, air/water ingress. Also, aerosol transport models for dry environments as opposed to human environments.

And I think earlier in the day you heard about the MELCOR development. That's addressing some of those issues.

So, our observations led to insights for follow-on work. The primary task that's proposed is the development of a guidance to support the staff's review of an HTGR PRA.

This guidance is intended to facilitate the understanding of where assumptions are likely to be made, what constitutes an adequate assumption, the relative importance of assumptions, the assessment of parameter uncertainties and also appropriate use of bounding analyses.

CHAIRMAN BLEY: You say the NUREG on uncertainty and LWR PRAs is something you need to expand?

Is that what these two bullets are suggesting?

MR. WOOD: Not expand, but just take consideration of the unique aspects of an HTGR design.

CHAIRMAN BLEY: When that came out, you said it was for LWRs.

MS. DROUIN: Can you repeat your question? I didn't quite follow it.

CHAIRMAN BLEY: You wrote a NUREG on uncertainty in LWR -

MS. DROUIN: And we're revising it. We're going to come see you in October.

CHAIRMAN BLEY: Oh. And that will now extend it beyond LWRs?

MS. DROUIN: No.

CHAIRMAN BLEY: So, my question was, is what's in there what you need to think about with respect to uncertainties for the HTGR, or are you having to go beyond the kinds of issues that you identified for LWRs?

MS. DROUIN: Well, you have to remember that that guidance is to - guidance to the licensee of how is he to assess these uncertainties in his PRA.

A lot of the process, and we said that before to the ACRS, a lot of that process is if you want to call it technology neutral, because it really doesn't get into the type of reactor. It gets into how do you treat the uncertainties in your basic events?

When you're calculating your risk measures, you know, how do you take those uncertainties? You know, how do you deal with the model uncertainties and how do you deal with all of this?

CHAIRMAN BLEY: How do you identify them all?

MS. DROUIN: I'm sorry?

CHAIRMAN BLEY: How do you identify them all? There's a lot of work on that provided.

MS. DROUIN: How do you go about identify them? That's in there, too.

So, a lot of that process would be applicable because it's really separate from the actual design of the plant.

CHAIRMAN BLEY: Thanks. That's what I wanted to hear.

MS. DROUIN: Oh, okay.

CHAIRMAN BLEY: Go ahead, Jeff.

MR. WOOD: Okay. Next slide. So, in parallel to this guidance development, we're also going to continue to prioritize activities to close any existing technical gaps. And this includes determining if there are any issues that require additional technical guidance development. And, also, to identify what the best vehicle is for that development. It may be an NRC-led task or industry-led or maybe some cooperation between the two.

And, also, we acknowledge that there are some issues that have ongoing related research activities that may be sufficient to address the needs for reviewing an HTGR PRA.

So, our status of this planning study, we completed the study. We completed our contract support in February. We have this study documented in a report.

Currently, we're coordinating with NRO to pursue follow-on activities which will be the development of the guidance that I discussed and also assessing any other activities that may be needed to close the technical gaps. And that's all I have.

CHAIRMAN BLEY: Thank you.

Anything more from the Committee?

Thanks very much. Next.

MR. BASU: Next is the human factor.

(Off-record discussion.)

MR. FLEGER: Good afternoon. My name is Steve Fleger. I'm a senior human factors analyst in the Division of Risk Analysis.

The objectives of our research were just to provide an overview of the plan for conducting confirmatory human factors research to support DOE's NGNP program.

That confirmatory research then that we're in the process of doing, will serve as a technical basis to support the update of the NRC's primary human factors documents - or guidance documents, I should say.

And there's three guidance documents; Chapter 18 of the Standard Review Plan, NUREG-0711 which is the programmatic guidance that is utilized for reviewing an applicant's implementation plan, human factors implementation plan or management plan; and then NUREG-0700 which provides a human factors guidance for reviewing the human system interfaces.

In terms of trying to scope this effort out, we're looking at the differences between essentially the existing Generation 2 plants, as well as the new Gen-3 plants and the advanced Gen-4 nuclear power plants.

These differences can be binned into two primary areas; the reactor design and the control room.

In terms of the reactor design, we're talking primarily about non-light water reactors that usually utilize passive safety systems and involve multiple modules.

The control room differences that we're going to see in the new and advanced plants are best described as being a highly-integrated control room.

And by that I mean they have digital I&C, increased use of automation, advance control algorithms. Usually involves sit-down consoles with soft controls and the use of computer-based procedures.

CHAIRMAN BLEY: Stephen, I asked this, I think, before you were here this morning, but we've just put it off until now: We've been looking at the new I&C and new control room designs for the plants that are coming in for design certification. As far as I can tell, they're working at the edge of technology right now.

Is there anything envisioned for the plants you're looking at that in any way go beyond what we're seeing in the new design cert plants?

MR. FLEGER: There is, and I'm going to get to that.

CHAIRMAN BLEY: You got that. Okay.

MR. FLEGER: Yes, yes. I'll get to that in a couple of slides.

The short answer is the SMRs. We're seeing some unique differences with the SMRS. So, I'll go over that briefly.

CHAIRMAN BLEY: Okay. Yes, we haven't seen any of that yet. So, go ahead.

MR. FLEGER: So, as a result of those changes in a highly-integrated control room, Research about maybe five, six years ago performed a study where we looked at emerging technology that is anticipated in these new plants.

And that report was summarized in a NUREG, NUREG/CR-6947, which I believe the same subcommittee evaluated earlier this year.

That particular NUREG identified 64

research issues as being new or different or that need to be addressed.

So, those 64 research issues, we divided them into about seven different human factors topic areas. And then we applied a PIRT methodology to bin those research areas into one of four priority areas and about 20 different topics were identified as being of the highest priority.

So, given the current state of knowledge, the differences between the new and the advanced nuclear power plants exist more with the design of the reactors and less with the design of the control room.

Therefore, the research that's needed to address human performance issues brought about by the emerging technology, are equally applicable for both new and advanced reactor designs.

So, most of the human factors research that's being planned or that's currently underway is applicable to both the new Gen-3 and advanced Gen-4 reactor designs with one exception. And that exception has to do with the modular reactors.

So, recognizing the differences, we sponsored a study about a year and a half ago to look at the human factors aspects associated with the concept of operations of modular designs.

And that project consists of four tasks. The first task was to develop a CONOPs model, and by that I simply mean a characterization of SMRs in terms of the concept of operations. How do the CONOPs for the SMRs differ from the CONOPs for the existing fleet of reactors, as well as for the new Gen-3 plants?

And so, that characterization I guess can best be described in a couple -- summarized in a couple of areas.

Typically, they involve small electrical outputs, 350 megawatts electrical or less, they involve multiple modules and have usually more than one mission in addition to power generation or maybe hydrogen production and process heat applications.

And probably the biggest area has to do with the area of operations from the control room operator's perspective. So, we're talking for the CONOPs for these plants, generally talking about operating -- a reduced crew operating from a single control room, multiple reactors which may be in a variety of states and running at different power levels.

MEMBER CORRADINI: So, I guess I don't understand the assumptions that you just said.

So, the thinking process is that there

would be a control room for multiple modules? And that links up with the current design concept from the - I'm thinking only the light water group, the light water reactor modules, or are you thinking beyond that to like NGNP-related that also there would be multiple modules in a single control room?

MR. FLEGER: No, so for the multiple modules, the multiple reactors, from what we've seen, it's looking like there will be just one control room.

And so for that one control room, they'll have a crew. And the crew may be less. We may have, you know, one operator for each reactor.

So, if you have a design consisting of 12 modules and 12 reactors, you'll have a significantly reduced crew all operating from one control room.

And so, the concept in terms of how do you lay out the I&C, how do you lay out the work stations which are digital, sit-down work stations? Do you have work stations for each reactor? Is that better? Or should you have just one work station and then have the operators all sharing that work station?

MEMBER CORRADINI: All there multiple module fossil plants that one can look at and see how they're designed now that would match up with that, or is there something unique about this relative to the module area?

MR. FLEGER: No, there are. And I'm going to talk just briefly about that next.

MEMBER CORRADINI: Okay. All right.

MR. FLEGER: So, given the CONOPs model with admission in the middle, we identified five supporting dimensions.

And so for each of those supporting dimensions, we looked at the - we're interested, I should say, in identifying the human performance issues associated with those dimensions which support the CONOPs model.

So, since there are currently no modular operating reactors, we had to look at surrogate systems.

So, through a combination of site visits, as well as literature review, we looked at - we examined offshore oil platforms, we looked at refineries, we looked at tele-intensive care units, naval vessels and -

CHAIRMAN BLEY: Did they share information with you?

MR. FLEGER: Yes.

CHAIRMAN BLEY: They did?

MR. FLEGER: Yes, they did.

(Off-record discussion.)

CHAIRMAN BLEY: Hey, you said you had this project.

Is there a report out on this?

MR. FLEGER: There will be this year, yes.

CHAIRMAN BLEY: Okay.

MR. FLEGER: I think the project has been going on for about 18 months now and it's currently slated to - the period of performance as far as I know, is supposed to end the end of next month.

CHAIRMAN BLEY: Okay.

MR. FLEGER: And then there's the review process. So, by the end of the year we'll anticipate having -

CHAIRMAN BLEY: Brookhaven again or somebody else?

MR. FLEGER: Yes, this is Brookhaven.

CHAIRMAN BLEY: We'd be real interested in following that when you get something.

Okay. Go ahead.

MR. FLEGER: Sean, I don't know. Are we going to be mentioning that on Friday?

MR. PETERS: Is this thing on?

CHAIRMAN BLEY: Yes, you're on.

MR. PETERS: Sean Peters. I'm the branch

chief for the human factors and reliability branch in the Office of Research.

And, yes, we will have a very small discussion on CONOPs, but we will have the author of the project, John O'Hara, in house. So, if you have any further questions, we can delve into that. At least the current state of that project.

It wasn't on the initial agenda to go into a detailed discussion of it, but we can expand upon that as we're closer.

CHAIRMAN BLEY: Super.

MR. FLEGER: Okay. So, real quickly then, the third task was to evaluate current NRC regulations and guidance to look at the suitability of that guidance for addressing human performance issues that were identified in the SMRs.

And for each of the six dimensions of our model that we developed, we did identify a number of human performance issues that are new that we feel need to be looked at.

The guidance review, NUREG - rather the regulatory guidance review is finished now. There were aspects of the existing regs that will require modification. And I believe there are also some new issues that we've identified, human performance issues that may require the issuance of new guidance or regulations.

And finally the fourth area then is to develop draft human factors guidance. And that will be done following our NRC's guidance development methodology.

In terms of our research to support new, as well as advanced reactors, it's been categorized into these six areas: The first has to do with research addressing human performance aspects of automation and human system interface complexity.

That study was completed last year. There's a technical report out on that. There's also another study underway that is following up on BNL's research that's being conducted by MIT. And I believe we're about halfway through the period of performance on that effort.

In the area of computer-based procedures, the NRC has been supporting IEEE's nuclear power engineering committee, Subcommittee 5's effort to develop new guidance on computerized operating procedure systems.

And we also have a task underway with INL to research some of the specific shortfalls we've identified in supporting the computer-based procedures standard.

CHAIRMAN BLEY: You've been up to the - any of the vendor facilities that are actually implementing those and see what they're doing?

MR. FLEGER: I personally have not been.

CHAIRMAN BLEY: I know NRO has been up.

MR. FLEGER: NRO has been.

CHAIRMAN BLEY: It's worth it. You learn a lot up there.

MR. FLEGER: Yes, I'd love to get up there. We do have people, though, real quickly, on the committee, IEEE committee, that's involved with the design of all that.

So, I think the standard will do a pretty good job at addressing some of the needed guidance for computer-based procedures.

CHAIRMAN BLEY: Well, I hope the people involved in the standards are actually getting to see what's being done, because there's some, I think, real surprises when you see what they're doing up there.

MR. FLEGER: Yes, there are a number of folks in NRO that are on that committee that have made those trips.

CHAIRMAN BLEY: Okay. Good.

MR. FLEGER: So, they are involved.

Human performance impacts of degraded I&C and HSIs, this study was also published last year. It's a BNL initiative. And as a result of that, we identified a lot of additional work that's needed.

So, in support of that, this year we'll be launching a new study to follow up on that work. And we can go into more detail on that, too. And, in fact, I know we are on Friday.

Fourth area, workload, situational awareness and teamwork. Got an SOW that's written and that, as I understand it, task was just awarded last month. So, that effort will be starting soon.

And then the considerations for staffing development and validation, that SOW is in process. And integrated system validation or the performance-based test contract will be also this year. And that's related to methods and tools development, which is an ongoing contract.

Some of our assumptions or research needs has to do with the lack of availability of the Gen-3 and Gen-3 plus designs in this country.

So, we're hoping to get some of the information from the Japanese and the French with the N4 and the ABWR. And I've just identified four bullets, four areas that we do need additional guidance on and hopefully we will be able to get from our colleagues overseas.

And the other area has to do with simulation and simulators. And although the applicants will submit research results needed to support their designs, it may be necessary for us to do some independent validation of that work. So, we feel that a research simulator will be essential to do that confirmatory analysis.

And then in moving forward, you know, we plan to continue our research. I'd say most of the research I just highlighted here in those six areas, probably half of it's done and the other half is underway with a couple of projects that will be launched this year.

That research then will be used to support the technical basis which will form the foundation for our primary human factors guidance documents. The first of which, Rev 3 of NUREG-0711, is slated to be released this summer.

CHAIRMAN BLEY: Can I take you back to your first bullet? The need for the simulator - I'm sorry. The previous page.

Is that something you see NRC doing, one of your contractors doing, something you do with

Halden or what are you thinking about?

MR. FLEGER: Well, probably all of the above.

We do have -- Research did procure a simulator. It's on board.

CHAIRMAN BLEY: Oh.

MR. FLEGER: It's on board. We have it.

CHAIRMAN BLEY: Down in Chattanooga or is it up here?

MR. FLEGER: No, it's up here. Church Street.

So, the plan is through a combination of those three, to do research in-house to utilize Halden. In fact, right now we're working on a bilateral agreement with Halden to support the integrated safety validation research that I mentioned.

CHAIRMAN BLEY: Okay. Good. Thanks.

MR. FLEGER: And that's all I have.

CHAIRMAN BLEY: Anything more from the Committee?

Stephen, thanks very much.

Sud, next.

MR. BASU: Next is I&C presented by Russ. We have two presenters also for this one.

MS. BANERJEE: I have only one for Dr. Yang.

MR. YANG: Yes, this presentation was prepared by both Russell and me, but Russ asked me to give the presentation so that I have some experience here.

CHAIRMAN BLEY: Well, welcome.

MR. YANG: But if you have tough questions, ask my boss.

(Laughter.)

MR. YANG: All right. So, I'm going to talk about NGNP instrumentation and controls. This is the scope of my presentation.

First, I'm going to talk about objectives of NGNP I&C research. Then I'm going to talk about unique I&C issues in NGNP and expected I&C design features in NGNP.

After that, I'm going to talk about three research projects in I&C, and what do we know from this project and what we don't know yet. The last part is our future plan.

First, our objectives is to provide technical basis for regulatory decisions and to develop regulatory infrastructure necessary to support the review of NGNP license application.

Next. In order to achieve our objectives, we need to know what are the differences between I&C - NGNP and I&C in light water reactor.

So, I'm going to spend a little time to talk about unique I&C issues which may affect I&C design.

The first thing is the harsh environment. You may know that NGNP temperature will be as high as like 570 degree, which is much higher than light water reactor.

We also may have -

CHAIRMAN BLEY: five, seven?

MR. YANG: 570 - thank you. 750 degrees.

CHAIRMAN BLEY: That's better.

MR. YANG: The light water reactor is below 400 degrees C. So, it's much higher.

We also may have corrosive materials if we use, say, SI processes in hydrogen generation where we have all the chemical processes.

Also, co-generation makes the I&C design different because now we are going to control not only the generator, but also we need to control the key processes, which makes the system more complicated. Will have more systems.

CHAIRMAN BLEY: So, if we have an attached hydrogen generation system or process system using the heat, NRC will be reviewing the I&C for that process system; is that true?

MR. YANG: We don't know that yet, but I think NRC is best to be prepared if we need to review both side.

CHAIRMAN BLEY: Thanks.

MR. SYDNOR: There are certainly the safety implications of that on the protection system.

MR. YANG: Because the - if hydrogen generation has some upset, it may require the reactor system to treat for the safety. That's why we need to consider it.

CHAIRMAN BLEY: That's fine. Go ahead. Thanks.

MR. YANG: Another issue is the advanced reactor design concept may affect the I&C design. For example, we now have clear cut capacity for graphite.

We also have reactivity design for fuel. We have a big margin between the operation temperature and the critical temperature because in normal operation, the temperature is 750 degrees. But the temperature to fill is 1,600 degrees. So, there is a big margin.

All this advanced design concept makes the dynamics much slower, which it gives the control system more time to respond if something is wrong.

And, also, selection of reactor design and heat process also affect I&C design. One example is like if we use pebble-bed reactor, then we need to measure the knob. But if we use prismatic, we do only this measurement. So, these are the main factors which may affect our I&C design.

Next. Besides these unique issues, we also expect I&C design has the following features. First, we expect the NGNP design needs more sensors, including unique sensors. By example, we may need moisture detector.

We expect digital I&C system will be used because it can efficiently handle the complex system. The system is much more complex than the AWR because we have an increased number of systems.

The digital I&C can effectively work with digitized sensors, digital data communication systems, computer supervisory control systems, FPGA devices, et cetera.

So, it also can use signal processing method to improve measurement accuracy and reliability. It can facilitate post-accident analysis because of the data log capacity. That's why we believe the systems will be used in the NGNP.

May not be used in protection systems, but it definitely - not definitely, but it most likely will be used in control systems.

We expect the digital control system will be able to coordinate all the subsystems such as reactor, steam generator, IHX, turbine, hydrogen production process work together with few human interactions.

We expect realtime monitoring and the diagnostics will be used for important parameters to improve system awareness.

Like this morning you may have heard that, for example, the crack detection we may need to have automatic monitoring systems to detect a possible failure.

We expect advanced resilient control including cyber-security, will be implemented to mitigate the consequence of unpredicted disturbance, sensor failures and malicious attack, et cetera.

Next slide.

CHAIRMAN BLEY: What do you mean by "resilient control"?

MR. YANG: Resilient control means if something goes wrong, the system can automatically reconfig so that the reconfigure system can still work, at least will not go to the next level. I mean the system will be safe. Will not go to the next level of -

CHAIRMAN BLEY: Have you thought about how we keep the operators knowledgeable about what their plant really is at this point?

MR. YANG: Yes, which means that we may need to educate the operators in a different way.

MR. SYDNOR: We don't know that these are going to be used, but we know that Idaho is looking at these features.

MR. YANG: Yes, Idaho had some report released last year, 2010. At the end of last year, had two reports suggest to use resilient control.

We have three research projects in I&C. The first one is advanced reactor control. The objective of this project is to review past and existing HTGR control design, track NGNP control design, provide technical information to NRC staff and develop regulatory acceptance criteria for advanced reactor controls.

The second project is advanced reactor instrumentation. We had a similar goal, but we emphasize our instrumentation instead of a control.

The third project is advanced diagnostics and prognostics. We have this project because we expect the online monitoring and diagnostics were the part of NGNP.

The objective of this project is to investigate issues arising from the integration of advanced diagnostics and prognostics system into nuclear power plant, including the impact on regulatory requirements.

What we know and what we don't know yet. First, what we know has been described in that report.

MEMBER RAY: This is the Rumsfeld version.

(Laughter.)

MR. YANG: First, the United States and other countries has accumulated a lot of experience on HTGR I&C design.

We have three NGNP pre-conceptual design released by DOE. So, this is pre-conceptual design by PBMR, by General Atomic and by AREVA, which described many different configurations.

Several hydrogen processes has emerged as leading processes. We know that DOE recommended one as the NGNP, but it hasn't been approved yet.

Several possible heat transportation systems are considered. And a few potential heat process applications are proposed.

From this information, we may infer the general requirements on I&C design, but we do not know the details yet.

So, that's what we - next, what we do not know. We will spend the rest of the time to investigate what we do not know and we will try to include those findings, you know, final NUREGs.

The first on the list of what we don't know is DOE hasn't made final decision to down-select reactor design. So, we do not know NGNP is a pebble-bed reactor or prismatic reactor.

As we previously mentioned, different reactor will have different requirement on I&C systems.

Second, conceptual design has not finished yet. So, we do not know what configuration will be NGNP. And we do not know exact I&C requirement yet.

Finally, engineering design will be after conceptual design if DOE decided to go ahead. So, we do not know the NGNP I&C design details right now.

Looking forward, we are going to track - continuously track NGNP development progress and follow the development of I&C system design.

We will analyze the licensing implications of NGNP I&C system design if the design will be fixed. We will update existing licensing guidance and develop new licensing guidance for NGNP I&C design as needed. That's it.

CHAIRMAN BLEY: You know, there's one thing that bothers me about the presentation. And that is it seems a lot like an advertisement for the advantages of advanced resilient digital I&C, and less a concern by a regulator about where this could get you into trouble in the future. And that concerns me a bit.

MR. YANG: We actually have reviewed all the - not all. Many of the existing guidance related to I&C systems.

And we found most guidance can be used for high-temperature gas-cooled reactor. That's good news. But there were a few ones, not many, to be updated or maybe we need to give up some new guidance.

CHAIRMAN BLEY: There are some things you had in the presentation about things you expect to see in this system that's taking digital I&C a couple steps beyond what we're seeing in the design cert plants like all the way back to the sensors and that sort of thing.

I mean, your colleagues in NRO are

struggling with how you make sure on the safety side there aren't any holes in these systems that can get you into trouble.

And if it's being expanded, it seems - I'd sure like to see more emphasis on that in this kind of a presentation.

I have to go back and look at the research plan. Maybe it's well done in there. I don't remember the details.

MR. SYDNOR: All of this is focused on impact of safety. And so, I think the presentation probably gave you that impression because we're stressing what might be, and a lot of this is yet to be approved, right, or DOE hasn't made decisions on.

All of these things are being looked at, may be considered, may be proposed. And so, I think the presentation stressed the unique aspects that we might be facing, because that's what we're focusing on with this research.

We're not trying to redo the licensing that's already been developed for digital applications in current and new reactors, because that's currently working.

So, this is focused on the things that might be problematic and could impact safety. So, I think that's why it gave you that impression.

It's not our intent. All of our research is focused on -

CHAIRMAN BLEY: When we get to the full Committee if we have anything on this, I'm not sure we'll have time to have this emphasize the other side.

And you've talked me into going back and reading more carefully over what it says about I&C and the research plan. Because the areas where it might be really new things, I think the research ought to be focused on just that issue of making sure you're not getting into any safety binds by having all the coolest things in the plant that you can have.

MR. SYDNOR: We're not proposing this. We're, you know, making sure that if it is proposed or if it is incorporated in a design that the staff is ready to analyze the impact on safety.

MR. YANG: Yes, exactly.

MR. SYDNOR: That's a better focus, and that is our focus. I think we gave you the wrong impression because, like I said, we were stressing what could be unique aspects.

CHAIRMAN BLEY: It all sounded like how good all these new things were.

MR. SYDNOR: Do I expect all of those to be implemented in? No, I don't.

CHAIRMAN BLEY: Anything from the rest of the Committee?

MR. BASU: As a regulator, we will not be designing I&C for the vendors anyway.

CHAIRMAN BLEY: I don't want to see us selling one either.

MR. BASU: Right.

MR. YANG: Yes, this expected aspects was actually based on what we have seen from the DOE-released materials from the pre-conceptual design material.

So, we thought that maybe that would be there. So, we better be prepared. It may happen or it may not happen.

CHAIRMAN BLEY: That's good. Thank you.

Anything more?

Sud, you're back up.

MR. BASU: I'm back. All right.

(Off-record discussion.)

MR. BASU: Okay. Now, this one is going to be a very short presentation for all the good reasons.

CHAIRMAN BLEY: Excellent.

MR. BASU: Not to mention that we all want to go home or go back to your hotel, right?

MEMBER CORRADINI: I'm trying to stay here as long as possible.

(Laughter.)

MR. BASU: And of course the professor has to be testifying tomorrow.

All right. No, but on a more serious note, on this particular topic we are on a standby mode. And you can understand why, because reactor design is not complete, as you heard the previous speaker, and process heat application demand has not been defined.

So, what I'm going to present to you is basically what I presented two years ago as the research plan. That still stands.

Given the Energy Policy Act assumption, which is hydrogen co-generation being the focus of the process heat application -

CHAIRMAN BLEY: Let's not dwell on that part much.

MR. BASU: Right. So, I'm not going to dwell on that. All I'm going to show you is the scope of research being if it was hydrogen co-generation, that we would be interested in the blast loading effect on reactor.

Of course for any process heat

application, we will be interested in the coupled thermal-fluid behavior providing the external loading on the reactor side and again from the safety aspect, component degradation issues again from the process heat products, and then the toxic and corrosive gas dispersion modeling.

One item that goes from the reactor side to the process heat side is the tritium migration that we will remain interested in.

As I said, we are in the standby mode. We haven't initiated any research in this area. We do have a wealth of data that have been accumulated through other programs in the past. Blast loading for one.

Then we go to the next slide. Decades of LWR hydrogen combustion research, we have blast models, combustion models. We also have decades of R&D on atmospheric dispersion modeling that we can make use of.

Material degradation and toxicology databases are out there for practically any toxic substance that you can think of from the process heat application.

If there are new toxic substances having an affect on the material degradation, we have to look into that.

We also have the benefit of some national/international programs. EUROPAIRS I mentioned. That is the end-use process heat that has started a couple of years ago.

There was some Korean and Japan R&D program on hydrogen. I'm not sure about their most current status on the nuclear energy university program funded by DOE and we should be able to benefit from that.

The DOE-sponsored activities that are listed here may be dated. They are from a couple of years ago and I'm not up to date on where DOE is with the down-selection of hydrogen production technology, whether or not that is still the major focus.

So, we'll have to just monitor the DOE program in that respect and see where that takes us in terms of the process heat application.

And then we'll revisit our R&D plan. And if there's any modification that we need to make, we'll do so.

So, with that I'll just mention one other thing. There are some administrative aspects that are not in the R&D plan for the right reason.

These are who has the regulatory

jurisdiction of the process heat plan? And I think you can relate to that.

And so, we have to have some dialogue with the other regulatory bodies ahead of time to make sure that NRC is on the same page with the other regulatory bodies insofar as the reactor safety is concerned.

So, if that means that we have to impose some requirements on the other side of the planet, we'll just initiate the dialogue with the right parties.

So, that's in a kind of nutshell in two minutes thing. I'm open for questions.

CHAIRMAN BLEY: From anybody?

Sub, thanks very much. And thanks, everybody, for the presentations. At this time, we're going to take a minute and go back over items we think we raised for questions that we'd like to hear more about.

And then we'll go around and hear from the committee members --

MS. BANERJEE: And the public.

CHAIRMAN BLEY: -- and the public. I would not forget them, but they come after the Committee.

Maitri, can you go down the list and we'll just see if we -

MS. BANERJEE: Yes, I compounded a list, but I may need a little help here.

CHAIRMAN BLEY: Yes.

MS. BANERJEE: The first one, I think multiple members had questions on how the staff is using the Fort St. Vrain experience and data and all that.

So, you'd like to hear little bit about that at the next presentation.

CHAIRMAN BLEY: Yes. I'm not sure we need that at the full Committee, but maybe the next time we talk.

MS. BANERJEE: Next time.

CHAIRMAN BLEY: Or feed something back to Maitri for us on that.

MEMBER REMPE: I think what I heard was the licensing experience was the way that they worded it. Not just Fort St. Vrain experience, but -

CHAIRMAN BLEY: That's true. The Fort St. Vrain licensing experience, yes, and how that's being -

MR. BASU: Okay. What I would suggest we do is we provide you with some information between now and the full Committee meeting. And at that point if you see any further need, we will -

CHAIRMAN BLEY: Okay. And be prepared in case it comes up at the full Committee.

MS. BANERJEE: And then your question on digital I&C, I think, went away.

CHAIRMAN BLEY: It went away.

MS. BANERJEE: Okay.

CHAIRMAN BLEY: I got answers on that.

MS. BANERJEE: And then how uncertainties are prescribed regarding the SCALE model development, you had some question there?

CHAIRMAN BLEY: I'll just summarize the idea. Uncertainty is clearly important in this area. Everybody talks about it.

That one slide bothered me and I just - we've recommended several times that the Agency have a complete process for - a systematic agreed-on process used everywhere for considering uncertainties and presenting them and identifying them and dealing with them, and that's the gist of my comment.

You're not going to answer that, but in the future we'll be pushing on that more. And that when uncertainty ideas are presented, they're presented in a coherent fashion and not things that aren't related to each other looking the same when they're presented.

So, the basic idea of what you're looking for in dealing with uncertainties, and then every analysis and presentation ought to be consistent with that approach.

So, that's not something to address now, but that's a long-term goal. And if we have - that might come up in a letter out of the full Committee meeting.

MS. BANERJEE: Like a generic kind of comment.

CHAIRMAN BLEY: Yes.

MS. BANERJEE: And you had a question on PARFUME. I think you wanted to see something, a technical report. Wanted to see something in the future.

MR. BASU: The user manual, the reference manual, those you are -

CHAIRMAN BLEY: Well, you have those manuals and we might want - I think we'd like to be able to see them.

The other thing was that Petti mentioned that there's technical reports and the possibility of presentations on the manufacturing process, measurement science and practice and process control, how that's all being done. And it sounded very useful to us.

So, at some point in the future like between now and the full Committee meeting, we would like to have a session where that's discussed in some detail and access to those documents.

MR. BASU: And my understanding is you will be looking at Idaho National Lab to give that -

CHAIRMAN BLEY: I think that would be most appropriate, yes.

MR. BASU: Right.

CHAIRMAN BLEY: Okay.

MS. BANERJEE: Okay. And then I'll get a copy of the ORNL document -

CHAIRMAN BLEY: Okay.

MS. BANERJEE: -- TM-2009. And then address graphite burning at the full Committee.

CHAIRMAN BLEY: Backup slides or maybe one slide that addresses it just to make sure that it gets on the table.

MEMBER CORRADINI: You might term it "graphite oxidation" since we've discussed what burning is.

CHAIRMAN BLEY: Well, there are - I think both are important to mention. I think we might have some vigorous discussion about it at the full Committee. I might be wrong.

I'd rather have it at the full Committee than have it in a letter-writing session.

MEMBER CORRADINI: Right, right.

(Off-record discussion.)

MS. BANERJEE: The RIM program, the ISI for graphite components, Harold had some questions and would like to be -

MEMBER RAY: Yes, I think the easiest way to express my concern is just show me that there is a explicit acknowledgment, don't just say it's part of some program, that long-term verification of the integrity of graphite structural members is part of the agenda.

I hesitate to use the word "in-service inspection," but that's what I mean.

CHAIRMAN BLEY: And related to that, Harold brought one up that I think everybody agreed with at the time.

In the first presentation, Sud had mentioned that there's nowhere in the current research plan that it looks at several key issues that need to be looked at eventually, including the fuel cycle safeguards and security in an integrated way in siting.

So, you know, we'll probably have a comment something related to that before we're done.

MEMBER RAY: The way I suggest, Dennis, or might think about it is that what we've reviewed here is basically the - I don't know what to call it - the research program or the path for pursuing these things that is defined in the six-year-old Energy Policy Act. And that was very specific. The plant was to be sited in Idaho. Period. Full stop.

And as long as that's a constraint, I mean, I think all we need to do is acknowledge it and say that that's what we reviewed here because it's very hard for both the NRC and the DOE to say, oh, well, we've decided to do something different. And especially for the NRC to do that.

So, given that that's the constraint, I guess you call it the roadmap or whatever it is that closed out of that, that the research is to satisfy the provisions in the 2005 Energy Policy Act for the NGNP. And that is very specific.

MS. BANERJEE: Is that a letter item or is that something you want to -

MEMBER RAY: Well, I would just -

CHAIRMAN BLEY: We'll talk about it.

MEMBER RAY: Yes, yes, yes.

CHAIRMAN BLEY: We won't decide that here.

MEMBER RAY: Obviously, but I'm just saying that that's the way I think about it. Rather than it looking like it's on a mission or something like that, it's just that the program is defined by the terms of the NGNP as defined in the Energy Policy Act.

And when it comes to siting, that's siting on a government facility as a demonstration plant in Idaho. That's it.

And I think we would all agree we need to acknowledge that, but not, like I say, indicate that it's somehow an omission or a flaw or something like that. It's just that's the boundary conditions for the research program.

CHAIRMAN BLEY: Maitri, anymore?

MS. BANERJEE: I think Sam had a question on densification of granular material.

MEMBER ARMIJO: No, it just was an observation that they had some nice experimental facilities. It might be able to be used to address the compaction of the core, a pebble-bed core quantitatively.

Shake those things and use that neat technique to see what the porosity value is.

MR. BASU: Yes, I mean, it got me thinking that. All we have to do is put that thing on a -

MEMBER ARMIJO: And just do it with your plastic things and see how your models are predicted. It's pretty slick.

MS. BANERJEE: And the last one -

MR. BASU: Maybe that's something we could suggest.

MEMBER ARMIJO: If I was back at school, I'd do it.

MS. BANERJEE: And the last one was the BNL project report on human performance CONOPs.

CHAIRMAN BLEY: We're going to hear more about that on Friday. So, I don't think we need that.

MEMBER CORRADINI: Can I ask just a general question?

So, from a standpoint of effort dose, 90 plus percent of all of NRC's effort is in the evaluation model and all the associated aspects of it in terms of where you call the R&D. And you're relying on, for all intents and purposes, all the preponderance, all the experimental data, et cetera, is in DOE's camp; is that fair?

MR. BASU: Well -

MEMBER CORRADINI: From a standpoint of

overall effort.

MR. BASU: Yes, you're right. Different sources. DOE, primary source in this case, but also applicant. The applicant's safety case.

MEMBER CORRADINI: But I'm talking about mainly from the standpoint of your effort, it really all centers around that picture that you or Joe, somebody showed early on in terms of the evaluation model.

MR. BASU: Correct.

MEMBER CORRADINI: Okay.

MR. BASU: Well, if you are just restricted to that figure, I don't know - I'll quantify as 90 percent, because remember now there is a bunch of research in the graphite high-temperature materials in those areas that are not reflected in the evaluation model.

MEMBER CORRADINI: But somehow it's going to be folded into it.

MR. BASU: Yes, we could call it a truth development for safety analysis performance verification, et cetera. And if we do that, then, yes, maybe 90 percent will be there, yes.

MEMBER CORRADINI: Thank you.

CHAIRMAN BLEY: Okay. At this time, I'm

going to go around to the committee members. And we've already hit some of this in the reprise with Maitri, but anything more you want to add?

Harold.

MEMBER RAY: Well, Dennis, I guess I just would say that looking ahead we need to - this is a hard letter to write. And I appreciate that's not the question you're asking me, but that's the way I'd like to frame the comment I want to make, which is what can we really say?

We've seen a lot, but it's at a very high level. And it will be at an even higher level at the full Committee.

So, I guess what we would be aiming at is, is there any omissions or something left out? But very hard for us to comment very much on the content of these programs because there are so many of them and we've only had, like I say, a high-level view of them.

It's inevitably the case, I think, that the NRC winds up doing work in parallel with the development that makes you ask yourself are we actually designing this or are we - we're not designing it. That isn't the word I want to use.

Are we doing the work that will then make it difficult for us later to be critical of what we've done looking at things like fuel performance and radionuclide models and so on.

And there isn't any way to avoid it, because we can't sit and do nothing. We have to do what's being done, but I would like to feel more clear in my mind about how we avoid becoming part of the thing that we then try and step back and look at objectively.

We talked about that a little bit somewhat today. But nevertheless, it's still on my mind. So, the upshot of it is I guess I would just want it to be clear that what we can say is limited by the information that's available and what visibility we have into it and the lack of definition at this point in time of what it is that will be used to evaluate.

So, there's a lot of qualifications to it, but mainly we would be just saying we do or we don't see that there's a gap somewhere, I would say, probably.

CHAIRMAN BLEY: Thank you. Sam.

MEMBER ARMIJO: Yes, I thought the presentations were very good. I learned a lot of this stuff from this work.

I think given the fact this is a

government-sponsored project and no private sector involvement to any significant extent, I just don't see that you can do anything other than what you're doing.

You've got to develop the tools and be sure they're - at least they're independent from what IO is doing. And I think you've outlined a very good set of programs.

I'm a fuel guy and I was very interested in what IO has been doing in the fuel area and some very nice basic work at University of Wisconsin. I have to admit that, Mike.

(Off-record discussion.)

MEMBER ARMIJO: So, I just think you're doing - with the limitations of this program, you know, DOE hadn't selected a plant design. DOE hadn't selected a fuel design. DOE hasn't selected a process - a heat - or how they're going to use the process heat.

So, within all those constraints, I think the NRC is doing the best they can and I think a very good job. So, I'm complimentary. I think it's a pretty good R&D program.

MEMBER RAY: Sam, could I just comment by saying that the Energy Policy Act says this is a partnership, not a government program.

So far, it's a government program.

MEMBER ARMIJO: I was talking reality, not the -

MEMBER RAY: But it's not meant to be viewed as a government project at the end of the day.

MEMBER ARMIJO: Well, that might be, but I'm sticking with it.

MEMBER RAY: All right.

CHAIRMAN BLEY: Mike, you came out of the other meeting very late, but I -

MEMBER RYAN: Thank you for the chance to join in late. Thank you.

CHAIRMAN BLEY: I know we have conflicts and things. So, within the range of what you two can say, Joy, is there anything you want to add or -

MEMBER REMPE: I just wanted to thank the staff for their presentations. I found them informative and appreciated them.

CHAIRMAN BLEY: Mike.

MEMBER CORRADINI: We're allowed to say anything, aren't we?

CHAIRMAN BLEY: You can say anything you want.

MEMBER CORRADINI: I guess the one thing I wanted to get back to is that I think in some sense just to reflect on - Joy found it, but I think it's important. In the revised draft program plan, there is a table in terms of what NRC views as theirs to be lead on and what things they're expecting from DOE. And I think that's important for us to see.

Other than that, I guess I wanted to ask Sud a question. Put him on the hot seat or Stu or somebody.

How far can you continue in this generic mode before you're presented with a design?

In other words, can you do this for another month? Another six months? Another year? Eventually all things are going to have to stop until you see a design.

MR. BASU: Eventually, yes.

MEMBER CORRADINI: So, can you define "eventually"?

MR. BASU: In terms of months, no. I'll be very hard pressed.

MEMBER RAY: Two years. Tell him two years.

MEMBER CORRADINI: Well, the only reason I'm putting it in that regard is I really do think we ended this - last time we had a briefing, I think

Sam's the one that said don't come back until you see a design.

And so, you're back.

(Simultaneous speaking.)

MEMBER CORRADINI: In some sense, I think that's where a lot of us will get very - will get more focused on asking questions.

MR. BASU: In fairness to Sam's remark, I think at that time our expectation was that in about a year, 18 months we will have - this comes down to a point design.

That evidently hasn't happened for, you know, a number of reasons. But the reality is we don't have a point design today. And your question is can I put my fingers on when we are going to have that?

The best I can answer you on that is whenever the Secretary is going to make a decision on -

(Off record discussion.)

MR. BASU: But this is not - this is not actually deterring us from carrying on our R&D at this point. We are in the midst of developing tools. We are in the midst of conducting experiments which are going to provide data for model development, code assessment, et cetera.

So, we'll get to some point which is still going to be somewhat generic in nature, but you're right.

At that point, we have to then either say, okay, stop the work because we don't have any further to go until we have the design. Or hopefully at that time we will have the design and then we can say, okay, now that we have this design, let's see what more we have to do.

CHAIRMAN BLEY: To me, the counterpoint to Mike's question is I think you've done a lot even designing a test facility now that you can address alternative design issues.

All right. Tom Kress, haven't heard from you.

CONSULTANT KRESS: Well, I have a number of thoughts. I'm probably out of - putting them down in a consultant's report, but I'll go over -

CHAIRMAN BLEY: I appreciate that, yes.

CONSULTANT KRESS: I'll go over a few of them anyway.

Number one, I thought the research plan was very thorough. I liked it. I think it covered all the bases. I thought it was really ambitious. I think you ought to be prepared to not meet all your milestones, and figure out what to do in case you don't.

And some of that data takes a long time to get. Particularly, I'm familiar with getting fission product release data. That takes an awfully lot of tests and an awful long time.

Some of my comments have to do with the white papers which I read before I came here as compared to the meeting.

I really liked the frequency consequence acceptance criteria that, you know, I've been an advocate for that for a long time. And I think for licensing basis, they've got the right idea.

They did carry it on out to beyond design basis. I had a little problem figuring out how their curve addressed the prompt fatality safety goal, but I'll work on that a little longer.

I could not convert - I didn't have Level 3, but I had ways to do it. So, I had a question about that.

I really didn't see any plans for steam graphite oxidation. I understand there are some, but I didn't see any in the plan. I think you will need that.

I think you should be prepared to have to provide some sort of technical justification for the assumption that the fission product species will be the same as we deal with in LWRs particularly because they have to pass through high-temperature graphite first where they have a chance to change.

For the fission product release models, the Booth-like models, I hesitate to think you can get effective diffusion coefficients for every fission product species.

I think you'll have to figure out some way to scale the other products from the cesium like they do with the LWR. I don't know what the scaling factors will be right now, but they may be the same as the LWRs.

I don't know of any models that exist for dust production through graphite. I would like to see what sort of models you're going to use for that.

And I didn't see good plans for how you were going to determine the amount of dust during operation and the nature of it, what its size and scale factors, shape factors might be.

MEMBER RAY: You think there should be a limitation on the amount that could accumulate, for example?

CONSULTANT KRESS: Oh, gosh, yes. And I think there will be a lot of it.

MEMBER CORRADINI: How are they going to know what that is in -

MEMBER RAY: Well, that's what he said.

CONSULTANT KRESS: I think it will be the driver for how much fission product release you get.

I don't think we know a lot about graphite plate-out and resuspension either, but those are going to be the drivers.

On the quality of fuel, I would like to rely on the standards for the manufacturing process to tell me my fuel quality meets the levels I want.

I would like to see for every batch that goes into the core, some sort of sampling and measurement of that core to that batch.

So, there's some issues about - statistical issues about how many samples, how you go about putting them in an in-pile and testing them and determining natural quality. So, I think there's some issues there that need to be thought about.

I have felt on pebble-bed reactors, I think spears won't move very nicely down through a pebble-bed. And there may be ways especially when they get irradiated and hot and the graphite changes dimensions, that you will get stuck spears in there.

And that releases to real hot spots and I think you'll need some sort of probabilistic model for these.

And I don't know how you deal with it in the MELCOR thermal models, but I think that needs to be given more thought and I think we might need some experiments.

I think I saw some where you look at the movement of spears down through a pebble-bed core, but I don't know if they were at the right temperatures. And of course they won't have the irradiation component to them.

I worry about the status of aerosol behavior graphite dust, but I've already mentioned that.

I, you know, one of the white papers said they're going to rely completely on mechanistic scenarios rather than design basis.

I really like the idea of design basis accidents. And I wish we'd think a little more about how we might develop those because, you know, you have to deal with uncertainties and the fact that you may be missing some scenarios.

You have to have defense in depth and I

think you deal with those well with design basis, as well as looking at the risk end of it.

And I'd be interested in this report that talks about how you use a PRA to develop licensing basis events that someone mentioned. I thought that would be interesting.

I was very pleased when I read some of the reports to see that site acceptance criteria, the SE curves, will be for all the modules taken together, not just an individual. I think that's necessary.

And because of that, I think when you talk about a site if we ever get to that point, the site has to be - you have to specify up front how many total modules you're going to have on that site. I think that's a policy issue that may have to be dealt with.

I think you have to think about common-mode failures with these. Particularly seismic and floods.

I think you're going to impact the model simultaneously with some things and I don't know how to deal with that in the PRA.

And I don't know what all the co-generation processes are, but I can't envision there should be any reactions particularly in intermediate heat exchanger and blast, toxic materials, things of that nature.

And the fact that the control room may have controls for that, I don't -- I would hate to see controls for the co-generation in the reactor control room, but I don't know how you're going to deal with that. I would rather see those separate.

Anyway, that's my initial thoughts on the subjects.

CHAIRMAN BLEY: A goodly list, sir. Thank you.

At this point, did you want to comment, Sud?

MR. BASU: No, I'm waking up.

(Laughter.)

CHAIRMAN BLEY: We'll send you the list.

MR. BASU: I'm tempted to respond to some, but I will, you know, when we get the list, we'll certainly respond. That probably would be better.

CHAIRMAN BLEY: We're now going to ask if there are any members of the public here in the meeting who have a comment or would like to say anything.

Is the phone line open now?

Please open the phone line.

Is there anybody on the phone line who would like to make a comment? I'll give you a second to make sure you're there.

(No response.)

CHAIRMAN BLEY: Okay. Nobody is speaking up. So, I guess there is not.

I would like to thank everybody for great discussions and good presentations. And I guess I ought to for the full Committee, I don't know for sure the schedule, but we'll either have only an hour, an hour-and-a-half probably at most. So -

MS. BANERJEE: I think they gave us almost two hours.

CHAIRMAN BLEY: Did they? Okay. I didn't see that. Okay. So, two hours. So, that's a small piece of what we did today. And you can't go over as many slides as fast because you certainly with the full Committee, are going to need gaps for allowing people to ask questions and dig into things.

So, you know, we can talk a little bit to make sure we get a package that's of the appropriate size. So, if we can see that a little early, it would be helpful.

At this point, thank you, everyone, and we'll call the meeting to a close. The meeting is now adjourned.

(Whereupon, the meeting was adjourned at 5:22 p.m.)

High Temperature Gas-Cooled Reactor (HTGR) NRC Research Plan
An Overview

Sudhamay Basu, RES
(sudhamay.basu@nrc.gov)

ACRS Future Plant Design Subcommittee
April 5, 2011

Outline

- Objectives
- Role and Scope of Research
- Assumptions
- Implementation Status
- Going Forward

Objectives

- Provide an update of NRC's HTGR Research Plan and its implementation
- Solicit ACRS feedback
- Request a letter from ACRS after the full committee briefing

Role of NRC HTGR Research

- Develop analytical tools and capability to:
  - perform confirmatory safety analysis
  - support licensing review
  - provide technical basis for regulatory decisions
- Develop technical basis for:
  - identifying and resolving safety issues
  - regulations and guidance
- Develop staff technical expertise and review capabilities

Scope of HTGR Research

- Confirmatory Safety Analysis Tools
  - Codes, evaluation models, data, V&V
- Major Technical Areas
  - Thermal-fluids, nuclear analysis, accident analysis
  - Fuel and fission products
  - Graphite and high temperature metallic materials
  - Coupling of reactor and process heat utilization plants
  - Structural integrity of systems and components
- Other Technical Areas
  - Probabilistic Risk Assessment (PRA)
  - Human Factors (HF)
  - I&C technology

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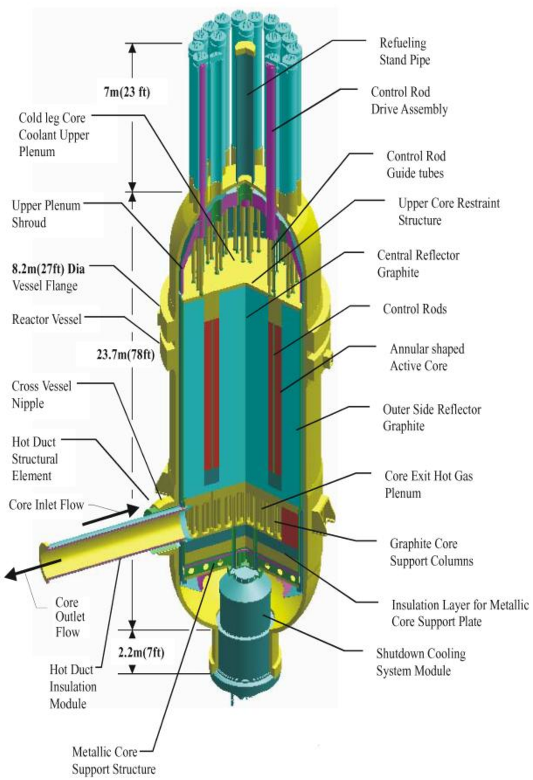
Assumptions

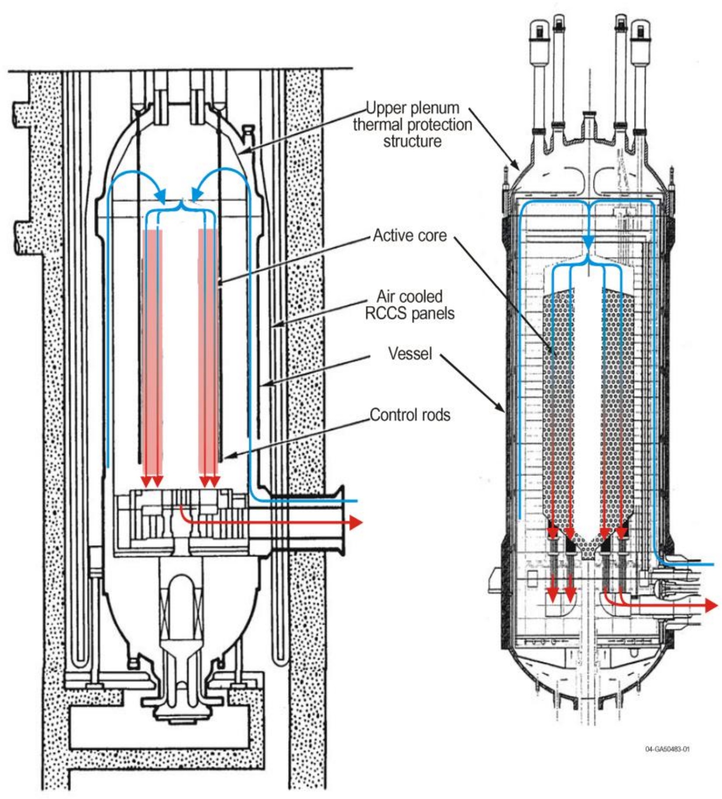
- Research scope in large part generic
- Availability of data from DOE-sponsored VHTR R&D
- Availability of applicant-furnished data for plant-specific licensing review
- Availability of complementary data from international HTGR R&D programs
- Reliance on national and international codes and standards
- Adequate resource allocation

New Development since Last Briefing

- Development affecting NRC R&D program
  - Decrease of reactor outlet temperature
  - Re-consideration of steam cycle for power conversion
  - Broadening scope of process heat utilization
- Other development with potential R&D, design, or regulatory impact
  - Co-location at industrial sites
  - Multi-module design
  - Consideration of fuel form
  - Consideration of technology alternatives

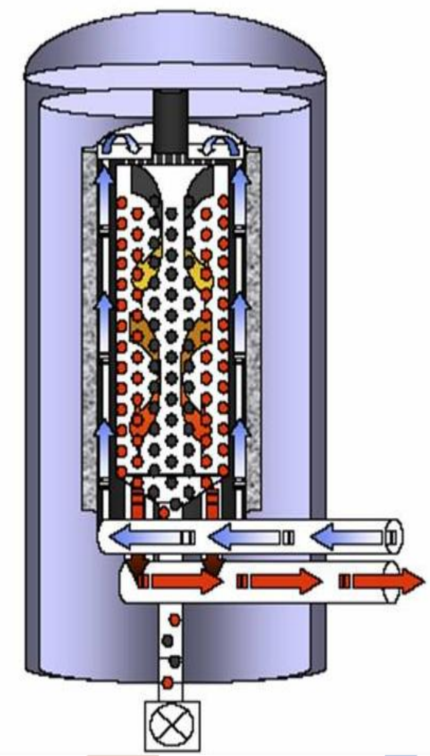
Prismatic and Pebble Bed Designs





a. Prismatic

b. Pebble-bed



Implementation Status
Thermal-fluids, Nuclear Analysis, Accident Analysis

- Thermal-fluid code and model development
  - MELCOR modifications for HTGR in progress at SNL
  - PARCS-AGREE development at U. Michigan
  - SCALE code suite modification and validation at ORNL
  - Developmental assessment using existing data
- Supporting experimental programs
  - High Temperature Test Facility at OSU
  - OECD-HTTR LOFC program (pending)
  - Core heat transfer and bypass flow studies at TAMU
  - Stratified counter-current flow experiments at PSU
  - Emissivity experiments at U. Wisconsin

Implementation Status

Fuel Performance and Fission Products

- Fuel fission product (FP) model development
  - MELCOR model for FP diffusion (coatings and matrix)
  - MELCOR model and code benchmarking
  - PARFUME code exercise and benchmarking
  - Metallic FP transport through SiC layer
- Fuel failure modeling
  - Mechanistic modeling of fuel particle failure in PARFUME
  - Empirical particle failure surface formulation for MELCOR
- Coordination with DOE/INL on AGR fuel program
- Regulatory guidance and oversight of fuel fabrication and quality assurance

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Implementation Status
Graphite and High Temperature Materials

- Properties and performance of graphite components
  - Stored energy release experiments and analysis at ORNL
  - Core component stress analysis tools development at ANL
  - Codes and standards activities
  - Coordination with DOE/INL on AGC program
- High temperature metallic materials behavior
  - Creep and creep-fatigue evaluation of RPV, IHX, SG, etc.
  - Develop time-dependent fracture mechanics methodology
  - Codes and standards activities
  - Coordination with DOE/INL on Materials R&D program

Implementation Status
Structural Analysis

- Assessment of concrete behavior at high temperature
  - High temperature effect on concrete physical and structural properties evaluated (NUREG/CR-7031)
  - Concrete strength generally retained within temperature range
  - Research on radiation effect on concrete initiated
- Seismic and Soil-structure interaction of deeply embedded structure
  - Previous study (NUREG/CR-6957) under review for appropriate update
- Seismic loading consideration for multi-modular design

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Implementation Status
Digital I&C and Human Factors

- Digital Instrumentation and Control
  - Research initiated on advanced reactor controls
  - Research on advanced reactor instrumentation
  - Investigate advanced diagnostics and prognostics (AD&P) system integration issues
- Human Factors (HF)
  - Developing technical basis to support update of HF guidance documents
  - Research focus
    - CONOPS for modular design
    - Automation and human system interface
    - Staffing consideration

Implementation Status
Probabilistic Risk Assessment (PRA)

- Planning study undertaken to identify PRA needs and scope for HTGR licensing
  - Gaps in guidance, methods, tools, or data to establish PRA acceptability
  - Assumptions for constructing a PRA model for HTGRs
  - Guidance development to support initial PRA review
- Other PRA related activities
  - Development of PRA Standards
  - Development of ANS 53.1 Standard
  - Research to close the technical gaps

Going Forward

- Continue focus on R&D that is generic to both reactor technologies
- Track DOE NGNP program and modify NRC R&D activities based on NGNP technology selection
- Continue coordination with DOE to resolve key technical issues and close R&D gaps
- Brief ACRS periodically on the progress

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Thank You

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NRC Evaluation Model Development for the Next Generation Nuclear Plant

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ACRS Future Plant Designs Subcommittee

April 5, 2011