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UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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583RD MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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FRIDAY

MAY 13, 2011

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Said I. Abdel-Khalik, Chairman, presiding.

COMMITTEE MEMBERS:

SAID I. ABDEL-KHALIK,

J. SAM ARMIJO,

SANJOY BANERJEE,

CHARLES H. BROWN, JR.,

DENNIS C. BLEY,

MICHAEL CORRADINI,

DANA A. POWERS,

(Continued)

COMMITTEE MEMBERS:

HAROLD B. RAY,

JOY REMPE,

MICHAEL T. RYAN,

WILLIAM J. SHACK,

JOHN D. SIEBER,

ACRS STAFF PRESENT:

EDWIN M. HACKETT, Executive Director

MAITRI BANERJEE

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

8:30 a.m.

CHAIRMAN ABDEL-KHALIK: The meeting will now come to order. This is the second day of the 583rd Meeting of the Advisory Committee on Reactor Safeguards.

During today's meeting the Committee will consider the following: One, Advance Reactor Research Plan; two, Future ACRS Activities Report of the Planning and Procedures Subcommittee; three, Reconciliation of ACRS Comments and Recommendations; four, Preparation for Meeting with the Commission; and, five, Preparation of ACRS Reports.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Ms. Maitri Banerjee is the Designated Federal Official for the initial portion of the meeting.

Portions of the session dealing with Advanced Reactor Research Plan may be closed in order to protect proprietary information. We have received no written comments or requests for time to make oral statements from members of the public regarding today's sessions.

There will be a phone bridge line. To

preclude interruption of the meeting the phone will be placed in a "listen-only" mode during the presentations and committee discussions.

A transcript of portions of the meeting is being kept and it is requested that the speakers use one of the microphones, identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We will now proceed to the first item on the agenda, the Advanced Reactor Research Plan, and Dr. Bley will lead us through that discussion.

MEMBER BLEY: Thank you, Mr. Chairman.

I'm Dennis Bley, Chairman of the Future Plan Design Subcommittee. The purpose of today's meetings 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 future NGNP applications.

On April 5th we had a meeting of the subcommittee with the staff on this subject. We were also briefly appraised of the related DOE R&D activity supporting NGNP.

Drs. Rempe, Corradini, and Powers may have organizational conflicts of interest with some topics and will not participate in discussions that involve areas of their conflict.

Today the staff is here to brief the full committee on the plan and I would invite Mike Scott to begin.

MR. SCOTT: Good morning. I am, as most of you know, Mike Scott. I'm the Deputy Director of the Division of Systems Analysis in the Office of Nuclear Regulatory Research.

I am here to introduce the staff, staff presentations and, of course, Sud Basu, to my left, who I'm sure you all know, will be the lead for today's discussions.

I regret I was unable to be here for the subcommittee meeting, but as I had mentioned to some of your members offline I was called over to Japan for two weeks after the Fukushima event, so I was unable to be present.

We are pleased to present to you today the research R&D plan, and we look forward to your feedback on that plan.

The plan has been submitted to the Office of New Reactors and the Office of New Reactors is going through their review process of that document and will ultimately, we believe, issue us a user need to direct the Office of Research in its continuing work regarding the NGNP R&D.

Of course, that sounds like the work got a little bit ahead of the formal plan and I think that would be an accurate statement, but the plan in draft form has been around for some time, and so the staff has been working on -- working to that plan, even though the formal plan just got approved in February.

DR. BASU: March.

MR. SCOTT: March 3rd. So, this is the plan that we've been working to and are continuing to work to.

NGNP is an important research activity with the staff. It consumes the majority of the current research advanced reactor R&D as distinguished from new reactor R&D.

We talk about new reactors in terms of the current generation, AP-1000 and other designs of that sort, and we talk about advance reactors in the next generation such as the NGNP, the IPWR and so on.

Because the IPWR design is somewhat less evolutionary than the NGNP, there has been a smaller identified need for research with regard to that design and to this one, the NGNP, hence the -- as I said, the NGNP is a major activity for the Office of Research.

Our view is that this is the body of work needed to support eventual review of the license application for the NGNP. There are always funding questions and those questions going forward could impact the schedule for our work on this but, again, we believe that this is the activities -- the activity list, the scope of work that's needed to get us to a position where we can adequately review the license application for NGNP when it comes in.

VICE CHAIRMAN ARMIJO: Assuming that you got the funding that you planned for, which never happens, but assuming that, roughly how many years before you really get 90 percent of what you're trying to achieve? Is this a five-year plan, a ten-year plan?

MR. SCOTT: I think probably five years is more accurate than ten years. Of course, some of the items finish a lot sooner than others.

Sud, what would be your answer to that question?

DR. BASU: Well, I'm going to give you a long answer. I mean, by --

VICE CHAIRMAN ARMIJO: Forget the money thing. Assume the money is okay.

DR. BASU: Yes. Assume the money is

there. By Energy Policy Act mandate we should be ready, in large part, by the end of 2013, and that was contingent on an applicant coming to us with an application by 2013. So that was the time they projected.

Right now we are looking at an applicant coming at the earliest, probably by 2015, so you can see that we have till then, even by the Energy Policy Act, if you will, but some of the research goes all the way to 2019, 2020 and so --

MR. SCOTT: But if you had no applicant, would you just stop work or would you just keep doing research?

DR. BASU: If we have no applicant?

MR. SCOTT: Yes. If DOE abandons the NGNP program then -- and we do not have any word, clearly, that they are going to do that, but if they were to do it, it would probably not make sense for us to continue spending resources on it because the Commission's guidance to the staff has been to focus on activities that are relatively near-term in licensing space.

And among advanced reactors, that's where NGNP fits currently. So, if it disappeared entirely it would be a very low priority in work with the

Office of Research. And, of course, with our customer, Office of New Reactors.

And with that, if there are no further questions for me, I'll turn it over to Sud.

DR. BASU: Good morning. Those of you who do not know me, my name is Sud Basu. I am the NGNP Research Program Coordinator in the Office of Research.

Today, my colleagues from the Office of Research and I are here to brief you on the NGNP Research Plan update and its implementation status.

We have given, as Dr. Bley pointed out, a briefing to the Subcommittee on Future Plans to Design on April 5th, so this briefing here would be a condensed version of the briefing that we have given.

And also, at the suggestion of the Subcommittee Chair, we are going to highlight to you three areas, and these are fuels, experimental work supporting the thermal hydraulics, thermal fluid -- analytical tools development, and finally, the graphite research.

I will, in my overview, I'll touch on other areas which are included in the R&D plan, but in a very brief manner. So by way of outline, I'm going to take you through the objectives of this briefing and the role and scope of research, some underlying assumptions that our research boundaries is as predicated upon, and then I will give you the implementation status of a number of research areas.

I'll conclude with a slide on going forward from this one onward.

So, the objectives is to provide you an update of objectives of the briefing. This briefing is to provide you an update on NRC's HTGR research plan and its implementation status, and to solicit your feedback and, of course, we will be expecting a letter from you at the end of the briefing.

The role of NRC's HTGR research is mainly to develop analytical tools and capabilities to perform confirmatory safety analyses and that is in support of licensing review and also to provide technical basis for any regulatory decisions that our regulatory office will have to make and undertake.

The role is also to develop technical basis for identifying and resolving issues and also to support the development of regulations, if any, certainly the development of regulatory guidance.

The scope of HTGR research, the plan that you have with you is to, again, do confirmatory safety and develop affirmative safety analysis tools and that includes development of codes, models, evaluation models, physical models, data, whatever data is needed and, of course, the validation and verification database.

The technical areas where we -- this plan is focused on, and that we will talk about today, thermal fluids, nuclear analysis, accident analysis. That data, as I mentioned before, there will be a presentation in that area by Joe Kelly.

We'll also talk about the fuel and fission products. Again, there will be a separate presentation by Stu Rubin, graphite and high-temperature materials. The graphite presentation will be given by Srini Srinivasan.

I'll briefly mention the work that we are doing in the high-temperature metallic materials area. It is the area of the process heat utilization. I have given the briefing before on that to the Subcommittee and full committee a couple of years ago.

I'll report on the status of that in this meeting. Also structural integrity of systems and components. There are other areas that were -- I will briefly touch upon, and that's in the area of probabilistic risk assessment, human factors, engineering and also instrumentation and control technology.

I mentioned assumptions that we made in developing the R&D plan as well as implementing the R&D plan, and the assumptions are listed here.

The first thing is that to date the research that we have conducted is mostly generic in nature, and by that I mean it's applicable to both reactor technologies, namely the pebble bed reactor technology as well as the prismatical reactor technology.

We are going to rely on the availability of data from DOE-sponsored VHTR program. That program's been ongoing for four or five years, and they are producing data and they will continue to produce data and we will rely on their data to both develop our tools as well as sensor tools.

We will also rely on the availability of applicant-furnished data, and these are data that the applicant will have to furnish to -- with their license applications to bring their safety case.

We will be relying on the availability of complementary data from international HTGR programs. There are a couple of programs ongoing, HTGR, high-temperature engineering test reactor program at JAEA in Japan and HTGR 10 program in China.

We had PB program previously. The last time we briefed you that program, unfortunately is not existent today. Of course, we have the program here at INL sponsored by DOE.

We will rely on national and international codes and standards in various areas and some of these staff will hopefully touch upon.

And, as Mike mentioned in his opening remarks, we will, of course, assume that adequate resources are allocated to -- from that research in a timely manner.

So, with those introductory slides, I am just going to go into the implementation status in various areas. And again, as I mentioned, Joe Kelly is going to have a separate presentation on supporting experimental programs supporting the thermal fluid code development activities.

I'm not going to go into that a little bit, for Joe to elaborate on. I'll just mention very briefly our activities in the code development, code monitor development area.

We are well into the modification of MELCOR code for HTGR applications, and there are more features that have been implemented into MELCOR. We are, of course, continuing to implement additional features into MELCOR so that we can use MELCOR for HTGR thermal fluid safety analysis.

That's going to be our main system level code but, of course, we will also have a parallel port to develop the couple PARCS-AGREE. That's the couple reactor physics code at the University of Michigan, and that code is basically going to complement the MELCOR code in various ways.

We are also modifying the nuclear analysis code scale for HTGR application and that work is going on at Oak Ridge National Laboratory and, as we are doing all this development work we are also conducting the developmental assessment using the legacy data, and using the data that is becoming available as we proceed.

In terms of the fuel performance fission products, again, Stu is going to talk about it, so I don't want to spend any time on this other than I'm just mentioning that Stu's presentation will cover fission product, and this is with regard to the fission product transport and release.

It will cover fission product release from fuel. There is also the other aspect of fission product transport which is in the primary circuit and in the reactor building containment, and that work is being carried out under the MELCOR development activity.

VICE CHAIRMAN ARMIJO: Sud, could you just go back one slide?

DR. BASU: The slide, sure.

VICE CHAIRMAN ARMIJO: The high-temperature test facility at Ohio State is -- what would they be testing there? Is it graphites, metal, what?

DR. BASU: That's a good question. In fact, we are in the midst of putting together an agreement with JEAE on the scope of work that could be conducted at the HTGR facility.

VICE CHAIRMAN ARMIJO: Does that facility actually exist?

DR. BASU: Yes.

VICE CHAIRMAN ARMIJO: That's nice.

DR. BASU: That's right.

VICE CHAIRMAN ARMIJO: I'd like to see American test facilities.

MEMBER CORRADINI: I think you're answering two different questions. He's talking about bullet two. You were asking him about bullet one.

VICE CHAIRMAN ARMIJO: Yes. Bullet 1. That's the one I'm -- yes.

DR. BASU: I thought, when you said HTTR -- HTTF, you meant?

VICE CHAIRMAN ARMIJO: HTTF, right.

DR. BASU: HTTF. That's at the Oregon State University.

VICE CHAIRMAN ARMIJO: Oregon State, not Ohio State? Okay.

DR. BASU: Not Ohio State.

VICE CHAIRMAN ARMIJO: And what will they test there? Is that an existing facility, will that be a new facility?

DR. BASU: That is for core -- thermal fluid core heat transfer bypass.

VICE CHAIRMAN ARMIJO: So, it's not a materials test facility, it's a --

DR. BASU: No, it's not a materials test facility. And Joe is going to elaborate on that in his talk. Sorry about that.

Okay. In the graphite area, as I mentioned, Srini is going to give a full talk on that, so I'm not going to touch on that.

In the high-temperature metallic materials area our purpose is to develop, again, tools to investigate the material performance under high-temperature conditions and in this case the focus is more on the creep and creep-fatigue evaluation of various materials that the reactor pressure vessel and heat exchanges, steam generators, et cetera.

MEMBER POWERS: The term up there, stored energy release experiments and analysis code language for radiation damage?

DR. BASU: That's an aspect of it and if you -- you have to hold that thought and question until Srini gives his talk, he will be able to elaborate on that in much more --

MEMBER POWERS: The one thing I know is Srini can elaborate on any aspect of graphite.

DR. BASU: Okay. Moving on then, in the structural analysis area what we have done so far is we assess the concrete behavior at high temperature. Well, this is an issue that has to do with the RCCS performance and RCCS integrity in case RCCS is not functional and during an accident the after heat of the decay heat is going to actually affect the integrity of -- will likely affect the integrity of concrete that surrounds the RCCS equipment, and so it is important to understand the concrete behavior at high temperature.

MEMBER POWERS: Are people thinking of using concrete suited for high-temperature behavior?

DR. BASU: I'm sorry.

MEMBER POWERS: Are people anticipating use of concrete that's suited for high-temperature environments?

DR. BASU: After the temperature that will be waged in case RCCS is not functional. So, if your question is are we going to develop or is someone going to develop concrete for high temperature, I'm not --

MEMBER POWERS: We don't need to. I mean, there are concretes where -- I mean, most of us think of concretes with a hydraulic bond, that there are certainly concretes that do not have hydraulic bonding and are admirable in their performance.

DR. BASU: Correct.

MEMBER POWERS: They are somewhat more difficult to cast, but they function well in high-temperature environments.

DR. BASU: Correct. In fact, that's the finding of the -- there's a report that came out, report on this work came out and the finding is that up to something like -- like 200 degrees C you are absolutely okay, and during the accident conditions for, you know, quite some time, you don't reach that.

So, with the concrete that is available

today you are okay.

VICE CHAIRMAN ARMIJO: Conventional. More or less conventional.

DR. BASU: Yes. You know, it's -- as Dana was saying, some special type of concrete that could be used.

We are starting to look into the soil structure interaction deeply embedded structures. Do you know that these plants are likely to be either fully or, in large part, embedded, so there's a lateral arc pressure issue and a soil structure interaction issue, particularly in the presence of seismic loading, so we are looking into that, and then we'll be -- you know, a longer term is to look at the seismic loading consideration for multimodular structure.

You first look at NGNP, it's not a multimodular structure, but eventually when commercial NGNP comes along that's an area that we'll have to look into.

We also see the equalization. There are a number of areas we identified as a result of our -- our phenomenon technique process and I listed them here plus loading, and that was the -- in relation to hydrogen core generation, thermal-fluid behavior of process heat components.

Of course, there is a mass and thermal exchange between the process heat side and the reactor side, so we'll need to look into thermal fluid behavior.

Component degradation issues, obviously, depending on what the process heat application is, you may have some toxic or corrosive byproducts that will have some impact on components, so component degradation issues are there and we identified those.

Of course, also the dispersing modeling of toxic and corrosive products, and finally the tritium migration modeling and that's a -- that's kind of a tritium migration from the reactor site to the process heat side so that it doesn't end up in the consumer products. That's the concern there.

We identified these issues. We really haven't started working on any of these. Pending better definition of what the process heat utilization will be for, you know, the first NGNP or any commercial follow-on thereafter.

The tools do exist in many areas from years of light water reactor research activities, so as of when the process heat utilization or application is defined in a better way, we should be able to utilize many, if not all of these tools to take care of these issues -- to address these issues.

Incidentally, the tritium migration modeling work is going on -- going on at Idaho National Laboratory, so we can also benefit from their activity.

In the area of digital I&C, human factors and PRA, in particular instrumentation and control we have initiated some work to survey advance reactor controls and instrumentation and also advanced diagnostic and prognostic issues.

You know, the reporting coming out, it's basically a preliminary survey of what's available today, what's good, what's not so good, what's applicable, et cetera.

In the human factors area, the focus is on development of technical basis to support update of the human factors guidance documents with the NUREG 0800, NUREG 0700 and 0710.

Let me see. What is the right number? 0711, and those have to do with the concept of operations automation and human system interface complexity issues and control room staffing and all those aspects.

In the PRA -

MEMBER BLEY: On that one -- I'm going to ask him not to talk much about this because there hasn't been a lot of work done, but in our session on human factors earlier, the last full committee, we had a more detailed briefing on that concept of operations idea, but they are really trying -- going to try to apply it here.

DR. BASU: Yes. Exactly. And again, in our focus at the moment is to update the human factor guidance documents.

And in the PRA area, likewise, we have undertaken a planning study to identify the PRA needs for -- and scope for the HTGR licensing and there is a draft document. I think we transmitted a document to you shortly after the subcommittee briefing.

We are also undertaking some other HTGR to look at PRA activities vis-a-vis coordination with the ANS 53.1 activities and NGNP white paper review.

Those are the purview of our new reactor office lead, so I'm not going to talk about those in this briefing, but you will have an opportunity to hear about those in future.

MEMBER POWERS: That would be very interesting because we essentially know plant operating plants, the database that you use for PRA must be really interesting.

DR. BASU: It is.

MEMBER CORRADINI: What did you say at the end? Must be very what?

MEMBER POWERS: Interesting.

DR. BASU: Interesting.

MEMBER CORRADINI: That's a word.

DR. BASU: And I can interpret it any way I like. Interesting.

It is sparse, but then, you know, we can -- we didn't license LWR based on PRA. We developed the operational data and the we applied PRA. So, perhaps some of the experience that we gained could be applicable.

Okay. So with that, I am going to just -- this slide going forward, we'll continue to focus on R&D. I mentioned that is generic, in other words, applicable to both technology until such time that DOE selects a technology and then we will refocus our effort in that technology at that point.

MEMBER CORRADINI: Sud, I have a question. I think I'm allowed to ask this question. So, when you -- when you say you -- it's not on your slide, but you said it -- that is we'll proceed forward when DOE has chosen its technology.

Define what you mean by "chosen a technology," because I think I know, but I want to be precise about this, since some members of another committee have emailed me about that.

DR. BASU: Okay. Chosen either pebble bed technology -

MEMBER CORRADINI: Okay. Fine.

DR. BASU: -- or prismatic technology.

MEMBER CORRADINI: Okay.

DR. BASU: And, for all I know, they may come back and say -- and I think DOE rep is here, so she can correct me, but they can come back saying that we are going to pursue both technologies. I don't know.

MEMBER CORRADINI: Okay. But if that -- if that last option were chosen, you'd have to have two efforts?

DR. BASU: If that last option, which is meaning going with both technologies, of course, all the work that we are doing now and will be doing until that point would still be applicable because we are generic in nature.

And then, of course, at that point we'll have to revisit the R&D plan to see whether we captured everything from both technologies in the R&D

plan and then also we'd like to proceed with both or if we didn't capture, then we'll revise that plan.

MEMBER CORRADINI: All right. Thank you.

MEMBER BANERJEE: Does your program coordinate with DOE's program?

DR. BASU: Yes. Very much so.

MEMBER BANERJEE: In what sense does that?

DR. BASU: Well, for one thing, any -- any topical area that I'm presenting here today, there's a program at DOE and we closely coordinate with them, our data, and make sure that it's generating or planning to generate serious data.

MEMBER BANERJEE: And what is your role then?

DR. BASU: What is my role?

MEMBER BANERJEE: I mean -- yes. What are you doing, more than DOE, or less.

DR. BASU: Okay. Maybe I kind of didn't state it. We are not generating any data by and large. With few exceptions, and these are the experimental programs that Joe is going to talk about.

Most of the data, in fact, all the data in the fuels area will be generated by DOE for other related programs. We will benefit from the data in terms of our tools development, tool success.

Like with the graphite area, DOE is probably going to generate it. And again, our tools development and assessment effort will benefit from the --

MEMBER BANERJEE: So those programs you showed, the experiments at OSU and so on, what are those? Are they DOE programs or your programs?

DR. BASU: Those are thermal fluids programs to start with in terms of area.

MEMBER BANERJEE: Okay.

DR. BASU: Ee started as an NRC program --

MEMBER BANERJEE: Right.

DR. BASU: -- but still we carried it in terms of funding and programmatic oversight, et cetera.

DOE subsequently joined that program and now it is a joint program. DOE and NRC program, and both DOE and NRC have oversight.

VICE CHAIRMAN ARMIJO: So, do you have a sort of overview of what you're doing and what DOE is doing and what the needs are? Is this sort of needs-driven work?

DR. BASU: Needs-driven work, and if you are referring to just the OSU program or --

MEMBER BANERJEE: No, no. I'm just

referring to everything because I'm sort of puzzled by what you're doing, what DOE's doing, how it meets the needs, what the road map is.

DR. BASU: We are -- in short, we are developing analytical tools to do confirmatory safety analysis and in the -- in the process of developing these tools whatever data we need and also in the process of development of assessment and validation, whatever data we need, most of that will come definitely in areas like fuels and graphite and high-temperature materials, those will come from DOE-funded work.

In thermal fluids area, a large part also will come from the DOE-funded work or international programs. And, in addition, we have initiatives for ourselves.

MEMBER BANERJEE: Is that because the other programs are not adequate?

DR. BASU: Well, I wouldn't -- I wouldn't quite characterize it that way. At the time we initiated some of these activities DOE did not have on its plate among these activities identified as such for immediate start.

VICE CHAIRMAN ARMIJO: Well, is that the Oregon State work? That maybe --

DR. BASU: Well, I think we are coming back to the Oregon State work, yes. Exactly, in -- for thermal fluids and again, Joe is going to elaborate on that in a lot more detail than --

MEMBER BANERJEE: Well, it's more than -- I'm looking for overview of how this program has come about, what are the drivers, where is it coming from and what are the needs being met, because do we know what the licensing needs are at the moment?

I mean, is this meeting --

MEMBER CORRADINI: I think it's in the report on Table 1.

MEMBER REMPE: The subcommittee --

MEMBER BANERJEE: I'm not a member of the subcommittee.

MEMBER REMPE: I know. The subcommittee got an NGNP research plan, and I -- the title of the document I have is "NRC NGNP Research Plan," but I don't have like a document number on it. Maitri sent it to us.

And so, there is such a document is what I would answer.

MEMBER BANERJEE: You mean a road map?

MEMBER CORRADINI: Yes. It has essentially the elements you're asking for. It has the needs and who's responsible, both between DOE and NRC.

MEMBER BLEY: And a couple years ago we had kind of thrown --

MEMBER CORRADINI: And we had a --

MEMBER BLEY: -- three years ago.

MEMBER BANERJEE: But isn't the whole thing sort of fluid, it's moving around right now as to what's going to happen? How are you able to develop a road map?

MR. RUBIN: This is -- maybe I can jump in here. If you want to look at a hierarchy of how --

MEMBER BANERJEE: Can you probably identify who you are?

MR. RUBIN: Oh, I'm sorry. I'm Stu Rubin, Office of Research. I apologize.

It started with the Energy Policy Act of 2005 and it laid out the role of DOE to basically be the developer of this technology for deployment in 2021, et cetera, and to provide R&D to support that.

It also defined the role of NRC, obviously, as a licensing authority, and also it defined NRC to interact with DOE to provide feedback on their own programs to make sure that what they are planning to do would be sufficient, if you will, in terms of the licensing application.

It also talked about the fact that what DOE developed in its R&D programs, both data and codes would be made available to the NRC, and so the overarching framework for our relationship at this time is defined in the Energy Policy Act, and that's the genesis of what we're doing now.

One tier below that was an MOU that we developed between NRC and DOE to further clarify what was in the Energy Policy Act and it talks about us providing feedback to them on the safety implications of what they're doing and the need to perhaps do more or different, and for them to take that on.

And I think I would venture to say that the Argonne facility was an area that we identified that we felt we needed to have more data to support licensing, and that was part of our feedback. And so they took that on.

But now, having taken that on, the Energy Policy Act says we're entitled to that data. We're entitled to any models that they may develop from it and so forth.

So, that's really the genesis there. It's kind of a unique relationship we have with them.

MEMBER BANERJEE: Yes. So they are in

charge of sort of promotion, right? And you're in charge of regulation. And I'm just wondering how that -- those rules are delineated.

MR. SCOTT: This is Mike Scott. They are clearly delineated in the documents that Stu is referring to.

I'd like to address your comment, Dr. Banerjee, about things are in a state of -- what you say, "fluid," --

MEMBER BANERJEE: Fluid, yes.

MR. SCOTT: -- changing. Yes, and things are always fluid, and if we waited for things not to be fluid we'd never have a plan, and so we decided to go ahead and put this plan in place, recognizing that it would be a living document and that as circumstances change, notably DOE's choice of a design, any changes in the project schedule and, as time goes on, we learn more, and that causes more questions which potentially changes the scope and the plan.

Yes, this will change. This is just to get us started down that road and we'll have additional road map changes as we go down the road.

MEMBER CORRADINI: I guess the only thing I would add to that is that Dennis remembers -- I think, as Dennis and John came on the committee, we had a letter that we issued in a previous version of the Research Plan which followed -- I'm trying to think, it was in 2007 or 2008, a PIRT process where they basically -- they, that is DOE and NRC together, came together and delineated this stuff.

Dana and I, in fact, participated in it, and in that they broke down things that were generic, regardless of technology, and things that were more technology-specific.

And I think, at least in the letter we wrote two years ago, three years ago, our thought at the time was -- the committee started, at the time was that, given a generic thrust, it was a good plan, at least for the moment.

MEMBER BLEY: And I'm sorry. We have a lot to get through in a short time, so I don't think we can go back all through that history here, but --

MEMBER REMPE: Just briefly. I hear, a lot of times, people talking about that when DOE takes a design and I think -- my understanding of the situation is that DOE wants an applicant to come in with their preferred design. And so, that's part of the issue about why there's not a design --

MEMBER BLEY: And that's certainly not

happening.

MEMBER BANERJEE: I think what I started asking is if there is an applicant in the horizon somewhere?

MEMBER CORRADINI: Always in the horizon.

MEMBER BANERJEE: Sud, I think you can go ahead.

DR. BASU: And, really, what I was going to say is that what Sanjoy asked is really the second bullet in this slide which is we're going to track the DOE program as we go along and we'll revise our program accordingly.

We'll continue to coordinate with DOE and, as your wish will come before ACRS subcommittee and the full committee to brief you --

MEMBER BANERJEE: How much effort is in this area that RES is putting in, roughly?

DR. BASU: In terms of dollar, FTE -

MEMBER BANERJEE: Oh, dollars, FTE or -- yes, just a few thoughts, roughly, what's going in there?

DR. BASU: My boss is going to answer you.

MEMBER BANERJEE: I don't know if you are allowed to or if you can, it would be helpful.

MR. SCOTT: I'll speak -- Mike Scott

again. Speaking for the NRC's piece of it, is that what you're asking?

MEMBER BANERJEE: Right. Right.

MR. SCOTT: In the Office of Research in FY 11, approximately 18 full-time equivalents are devoted to this. Now, that's not -- you know, that's people managing the projects as well as people doing the research.

The contracting, it's on the order of millions of dollars. I don't remember the exact number. We are subject to substantial budget pressures and where this goes over the longer term, we'll just have to wait and see.

DOE has a much larger budget, of course, than we do for their R&D.

DR. BASU: Okay. That does it and the next presenter will be --

MR. RUBIN: Okay. Okay. Stu Rubin, Senior Technical Advisor, Office of Research. What I'll be discussing is the fuel performance and fuel fission product transport R&D arena.

I'll start with a very quick overview of DOE's advanced gas reactor fuel development and qualification program, and I'll discuss how that program is supporting the NRC's R&D needs in the area of fuels analysis, as well as accident analysis modeling and data needs.

I'll discuss INL's PARFUME code and how we plan to further develop it and use it in a regulatory sense in the application.

I'll discuss the new HTGR fuel models that we're programming into MELCOR, so MELCOR can calculate fission product release from an HTGR core during normal operations and accidents, and I'll touch upon some of the activities we're involved with to develop our staff so they better understand the fuel technology, and wrap up with some guidance that's been developed in the area of inspecting fuel fabrication facilities.

As far as DOE's advanced gas reactor fuel development qualification programs, this slide presents it basically all in one slide. It consists of eight irradiations, AGR-1 through 8, shown by the orange boxes.

Each irradiation is followed by a safety testing or heat-up testing, as well as post irradiation examinations, and those are shown in the red boxes. So each irradiation has a different purpose and those are shown in the yellow boxes.

As far as AGR-1 is concerned, which is

irradiation's now complete, it will involve fuel compacts with coated particles with UCO kernels that were -- had coating layers that were fabricated at what we'd call laboratory scale, using a German type coating process and it also includes particles with small variance on the coating process.

The primary purpose of AGR-1, however, is to shake down the test capsule design so make sure it's going to work effectively in the later irradiations, such as the fuel qualification of radiation

AGR-2, which is now under radiation is basically a fuel performance demonstration irradiation, involves UCO and UO2 fuel particles made with a much larger industrial scale coater similar to what will be in a production facility. It also involves UO2 particles that were provided by pebble bed in South Africa and UO2 fuel in compacts provided by CEA in France.

And the purpose of AGR-3 and 4 are basically to obtain the data that's going to be needed to model fission product transport in the NGNP fuel form so as to develop the action analysis term capability.

And it's going to involve compacts that

include particles that are designed to fail very quickly in irradiation and to put out a well-defined quantity of fission products over the entire radiation, and those, then, will allow for a cleaner analysis at the back end to -- a back out to diffusion coefficients.

AGR-5 and 6 are the formal fuel qualification tests, and the objectives there is to demonstrate fuel performance or fuel particle integrity as at a high rate at the NGNP design service conditions. And there will be accident condition testing in that as well.

And finally, seven and eight are tests that are designed to allow for validation of both codes that are -- can predict particle failure as well as codes that are to predict fission product release from the fuel.

I'd now like to move into an important part of the AGR program that is not mentioned previously, and that is the code that was developed by INL over perhaps the last ten years, and they are now using in connection with their AGR fuel performance -- or fuel program for not only fuel design and development, but also in terms of pretest calculations for each of these AGR test areas.

The coat is called PARFUME. It stands for particle fuel model. It could be characterized as the state-of-the-art TRISO particle fuel performance analysis code.

It can calculate both the failure probability of fuel particles in compacts or pebbles, as well as the fission product release from the fuel compacts or fuel pebbles.

It predicts the failure probability of particles by basically analyzing the thermal, mechanical and physiochemical state of the particles over time and then compares that state at each time with a failure probability curve, a liable type of failure probability distribution versus stress.

It does model many of the important phenomena that it identified it can lead to particle failure such as internal pressure, cracking of debonding of the paralytic carbon layers, radiomigration of the kernel toward the -- toward the coatings, increased local stress due to out-of-roundness and so forth.

It also has the capability to account for statistical variations in the thicknesses and properties of particles due to manufacture, and to factor that into the -- into the analysis.

Again, as part of the MOU, which is again a part f the EPACT 2005, NRC is able to obtain this kind of a code from DOE and INL and also the data that supports it, and we have taken advantage of that and have recently obtained that code in December of last year and are now starting to work with it.

Quickly, this is an outline of the modeling of PARFUME, the continuity equations, if you will. Again, it solves the temperature -- temperature profiles in the fuel element as well as the spherical pebbles or particles, themselves.

One of the unique and powerful parts of it is shown in red. It analyzes the stress/strain state in the particles and then goes through a failure probability analysis.

VICE CHAIRMAN ARMIJO: Is that for just totally spherical or does it allow the particles --

MR. RUBIN: No. It --

VICE CHAIRMAN ARMIJO: -- out-of-round and --

MR. RUBIN: It does account for the things that I described in the previous page, out-of-roundness -- let's say bends in the layers --

VICE CHAIRMAN ARMIJO: Bent layers.

MR. RUBIN: -- things of that sort. It

accounts for all of that.

MEMBER BANERJEE: So, how does it do that? So you don't assume spherical symmetry?

MR. RUBIN: You can. You can.

MEMBER BANERJEE: That's the idea, right?

MR. RUBIN: That's the idea, but the way particles fail is details of the distribution that dominate --

MEMBER BANERJEE: All right. Let's go back to how you solved the equations here.

MR. RUBIN: Well, I -

MEMBER BANERJEE: Is it in -- let's -- first order question. Does it assume directly symmetric or do you actually utilize the whole particle in an arbitrary --

MR. RUBIN: Well, basically, the layers and the kernel -- the layers, themselves, are finite elements --

MEMBER BANERJEE: First of all, are you guys handling this or DOE?

MR. RUBIN: No. No. Again, they have spent their own funds and DOE funds to develop the code. It's reached the point now where it's an operational code. It's had some benchmarking as well as now some benchmarking at SCHER, and we feel it's --

would be a valuable code for us to now have, but there's more work that still needs to be done.

MEMBER BANERJEE: So it takes into account variabilities, out-of-roundness --

MR. RUBIN: All that.

MEMBER BANERJEE: And how does it do it? Is it a finite element code or --

MR. RUBIN: Fundamentally, in terms of the stress/strain analysis, the finite element code --

MEMBER BANERJEE: But the production -

MR. RUBIN: Now, I'd be doing a disservice to try to explain it simply. There is a theory manual that we can provide. It gives you extensive detail on that, but basically you model the roundness or out-of-roundness that you want in the basic finite element model that --

MEMBER BANERJEE: Well, everything within a finite element structure?

MR. RUBIN: Yes. The stress/strain calculation --

MEMBER BLEY: And somehow must be treated probabilistically that you can have --

MR. RUBIN: Yes.

MEMBER BLEY: -- variations in all those --

MR. RUBIN: The probabilistic overlay of variations in thicknesses --

MEMBER BANERJEE: But samples from a distribution of some sort?

MR. RUBIN: Yes. You can -- you can essentially put in the actual manufacturing distribution that you measured from a particular batch or batches and the variation of that, and put that into PARFUME and say, "This is the variation of the particles I want you to do this statistical analysis on."

And again, it's details in going through the sampling of those various parameters that --

MEMBER BANERJEE: Well, that's -- yes, that's a separate issue. I'm just after how you solved it because, in order to be able to solve the problem you have to be able to have a variation in the thicknesses and do all the calculations and how are the interface resistances put in.

I mean, it seems like it's not obvious.

MR. RUBIN: Yes. Yes. I think that probably deserves its own meeting to get into that.

MEMBER BANERJEE: Right.

MR. RUBIN: And it's not --

MEMBER BANERJEE: But it's not your code.

It's DOE's code.

MR. RUBIN: No.

MEMBER BLEY: Let me try to ask Sanjoy's question in a very naive way. It sounds like you're saying these variations are all down, down at the deep level of the model and then recalculated. It's not some kind of overlay, fudge factor distribution to account for these variations.

MR. RUBIN: Right.

MEMBER BANERJEE: Thank you.

MR. RUBIN: I'm looking at now --

MEMBER BLEY: That's kind of what you're --

MR. RUBIN: -- DOE and I know, and the people who have been trained in our code to get down to that level in terms of the solution approach to the -- overlaying the variations to the basic finite element, and I personally don't have an answer for you.

MEMBER BANERJEE: Yes. I think what -- Dennis put it well but, you know, there are many ways --

MR. RUBIN: Yes.

MEMBER BANERJEE: -- to do this, but if you are actually taking into account the distribution of thicknesses in some way which is being sampled from a Monte Carlo or something --

MR. RUBIN: Sure.

MEMBER BANERJEE: -- it's very different from just arbitrarily doing this with a fudge factor afterwards.

MR. RUBIN: Let me jump to the next slide.

MEMBER BLEY: Let me -- excuse me just a minute before we go on. I was just reminded and I'd forgotten this.

From the subcommittee, we had requested to actually have a briefing on PARFUME at a later date.

MR. RUBIN: Okay.

MEMBER BLEY: So maybe that --

MR. RUBIN: That's what I would suggest, and I think we've given you the theory manual on that, and I would guess that DOE and INL certainly participate, the developers in that.

But let me jump to this slide, and this slide depicts how we're going to -- and I know, and Harold is going to further benchmark, develop and ultimately, hopefully, validate.

And let me go to the last row -- the last row here is AGR-7 and 8, and what that is intended to do is to provide test data on particle failures. The test will be run in a way to drive particles to fail.

The previous tests are now expected to have significant -- and so you will see from the actual data and the variations from manufacture, et cetera, and the conditions you impose, what those failure times are, and that will give you a validation data set to then compare to what PARFUME or other codes might predict.

MEMBER SIEBER: Can you tell us what a failed particle really is?

MR. RUBIN: A failed particle, for me, is defined to be a particle where all of the coating layers have cracked and there is a -- all of them.

MEMBER SIEBER: Or just a path through?

MR. RUBIN: There's a crack and so there's an escape path --

MEMBER SIEBER: Right.

MR. RUBIN: -- for gaseous and metallic fission products to quickly short-circuit out of that particle.

MEMBER SIEBER: Okay.

MR. RUBIN: All right. Just for the sake of time, let me move on.

This is basically the matrix of tests that we're going to tie our benchmarking development and validation to. That last one is a key for us and others.

In terms of the planned uses of PARFUME, this slide depicts that it's not formally part of the evaluation model for -- that the staff is developing, but we do put in the engine as a stand-alone code.

One use we anticipate is to use it to assess the applicant's failure model that they are going to use in our evaluation mode. It will also allow us -- I think it is very good for this to perform sensitivity studies.

Maybe the absolutes may be off and we don't know, the benchmarks will tell us, but the sensitivity part will be a valuable tool to assess variations in manufacture of lot-to-lot and so forth to see the effect of particle failure as well as variations in operating conditions.

VICE CHAIRMAN ARMIJO: Will the applicants have access to PARFUME? Will they be able to use it as well? I would think that would defeat the purpose.

MR. RUBIN: I'm not aware of any of the applicants that we understand are interested in the NGNP project are looking for that.

VICE CHAIRMAN ARMIJO: They have their own?

MR. RUBIN: Well, the South Africans, when they were a player, were taking an approach where they wanted a strictly an empirical approach to particle failures that we're using in the safety analysis.

GA as a model, it's an empirical model, but it's driven by conditions of the particles, so it's an empirical fit for different failure mechanisms. Okay.

The other -- the other source are the French, and they have a code which escapes me. It's very much like PARFUME, and I can't think of the name. It is a mechanistic code.

So, if you look out at the lay of the land you don't see anyone who's saying, well, we like this, too, but they could. Okay. Let me just keep going. Now --

MEMBER BLEY: Before you leave PARFUME --

MR. RUBIN: Sure.

MEMBER BLEY: -- and I know we're going to have a briefing on that later, and you said you're going to ask INEL and DOE.

MR. RUBIN: Absolutely.

MEMBER BLEY: I might be a little bit old-fashioned, but it strikes me as NRC is going to be using this, you guys ought to understand the guts of the workings as well as the developers by the time you're using it.

MR. RUBIN: If you look at the fuels R&D plan, it is essentially what you are talking about. We are going to use it starting by going back to some of the benchmarks that were done for an IAEA CRP and do it ourselves and see if we can get the same answers.

We also had someone in training, by the way, for five days who got the theory part of it. We're then going to march along in lockstep with the AGR test theories, doing pretest and post-test calculations with PARFUME to compare with the actual test results and also with INEL's.

MEMBER BLEY: So at some point in the future you do intend to have this deep understanding of --

MR. RUBIN: Oh, that is a key, is to --

MEMBER BLEY: Thank you.

MR. RUBIN: -- get a profound understanding and be a user that understands the code, for sure.

MEMBER REMPE: Well, isn't there also an effort underway to take key aspects of the model and put it in MELCOR on the NRC --

MR. RUBIN: No.

MEMBER REMPE: Oh, I thought you were going to have some sort of correlation in MELCOR to predict failure and so it would be the NRC approach would be using MELCOR -

MR. RUBIN: Again -- again, PARFUME is a mechanistic code. It has -- it has to go through a solution of these field equations to get at a particle failure rate.

When you look at the time involved --

MEMBER REMPE: Right.

MR. RUBIN: -- it's just not practical.

MEMBER BLEY: No, I think what she was saying --

MR. RUBIN: So, what we're doing is, we're taking an empirical approach.

MEMBER BLEY: Okay.

MEMBER REMPE: Right. And putting that into MELCOR, right.

MR. RUBIN: That's going to be put in MELCOR, yes. Absolutely.

MEMBER BANERJEE: I thought, did you start a model in if you wanted to?

MR. RUBIN: In fact, that shows up, if you look at the circle on the lower left, we talk about a failure fraction, that's an empirical -- and it's a data set.

MEMBER REMPE: That's what I thought your approach was --

MR. RUBIN: The temperature I'm at, oh, I'm at this temperature, I go -- well, this is my failure fraction.

MEMBER REMPE: Right. So that's what I -- although you -

MEMBER SHACK: But your slide four has -- says that, you know, MELCOR will have everything except that failure model, and then you will use --

MR. RUBIN: Yes.

MEMBER SHACK: -- an input/output fraction curve or --

MR. RUBIN: Right.

MEMBER SHACK: -- response surface.

MR. RUBIN: Right. It's a fraction -- virtually -- I think all of the applicants are using some kind of an empirical approach to predict. It's not a mechanistic approach. It's just hard to validate, among other things.

Okay. Now, again, this is our evaluation model. I'd like to focus in on the two boxes on the lower part of the curve, which basically are MELCOR or the key there, and I've circled the pieces where the HTGR fuels R&D is supporting the development of models and the benchmarking of models and the validation of models within this scheme. Okay. Now --

MEMBER BANERJEE: Just one thing. If you do this process you want to be able to propagate uncertainties in though this in some systematic way.

MR. RUBIN: Correct.

MEMBER BANERJEE: Because, this is an inherently multiscale problem you are trying to do.

MR. RUBIN: Correct.

MEMBER BANERJEE: And it's not obvious how you are going to do that uncertainty propagation. I'd think about it --

MR. RUBIN: Yes. We have --

MEMBER BANERJEE: -- if you haven't already.

MR. RUBIN: -- and we have developed a statement of work and have a contractor in place whose function in the statement of work includes that whole -- that subject specifically, is develop a scheme to account for uncertainties in all piece parts of our evaluation model and to kind of propagate that to an overall uncertainty in the figure of merit which, for us, is the releases or dose, but all these things --

particle failures, there's a statistical variation there. That's one piece for sure, but there are many, many others, and we do have a contractor working on that whole question of how to handle the uncertainty analysis and actually make it work.

MEMBER POWERS: It's probably fair to --

MR. RUBIN: Joe Kelly is our leader for that effort.

MEMBER BANERJEE: Dana, you were saying something?

MEMBER POWERS: Well, if I may just point out that endemic to the design of MELCOR is that any parametric quantity is available to the user and they have devised the algorithms and, indeed, the coding to conduct Monte Carlo sampling of whichever set of parametrics you want in the code, to do a Monte Carlo type uncertainty analysis, and it's, quickly, how many processors you want to tie up doing calculations, is the limitation on that.

MEMBER BANERJEE: Does it include the model uncertainties?

MEMBER POWERS: Well, modeling -- no. These are strictly parametric uncertainties. Modeling uncertainties, you know, are the grand imponderables. I don't know what to do about them.

MR. RUBIN: I'm going to have to make up time here, in going through my slides, I could tell. This is a --

MEMBER BLEY: Well, you finished ten minutes ago.

MR. RUBIN: But that wasn't your fault.

MEMBER BLEY: Yes, it is.

MR. RUBIN: Okay. Here is -- this is a MELCOR core model. I think it -- I believe it's for a pebble bed type reactor. I've highlighted an HTGR fuel calculation unit cell. It's a single one of those red blocks defined as the active core, and there have to be enough of these -- there has to be a fine-enough mesh so that we can pick up the small differences in temperature and burn-up of fuel in each location because the -- both drive the particle failure rates and certainly temperature drives the diffusion rates, even for intact particles, so it is a fairly fine mesh.

Now, within each unit cell we now have to model the releases from fuel. This is a comparison, side-by-side between prismatic fuel element, core model or unit cell made and pebble bed. For the most part they are the same.

The differences lie -- in the prismatic

there is a gap between the compact outer surface and the inner surface of the fuel bowl that contains it and then, of course, there's the graphite block that needs to be modeled in the -- in the transport. Basically they are the same, though.

Now, in terms of initial results in putting in some of the modeling into MELCOR, what this represents is a results of a benchmark from an IAEA CRP-6.

These are the results of the participants. MELCOR's highlighted in red and what we have here is a series of benchmarks. They are basically theoretical constructs. They are defined in each case on the lower part of the curve.

And so, in comparing the MELCOR predictions of cesium release after so many hours at such a temperature for these various type particles, you can see there's a good correlation or a good consistency there between the two codes and I would like to think that that kind of shows that we're putting the models in correctly, and we are verifying in a way that the models are now coded properly.

So, that validation, you know, in some cases we do have a theoretical answer and we're consistent with the theoretical answer.

CHAIRMAN ABDEL-KHALIK: Code-to-code comparisons are often misleading. I mean, these are all codes that are compared against each other.

MR. RUBIN: Yes. It's a code-top-code.

CHAIRMAN ABDEL-KHALIK: They can all be wrong.

MR. RUBIN: Yes. Now, there are -- not shown here -- there are some additional benchmarks we haven't done yet which are actual experiments. We have not had the time to go back and catch up with the CRP benchmark to see.

Now, there, as well, I'd have to say the predictions were off in a number of cases by orders of magnitude from the actual measured releases.

MEMBER CORRADINI: Right. But I guess I -- I guess I interpret Said's point -- I guess I interpret Said's point is that if PARFUME is your mechanistic tool it, in some sense, has to be validated based on those experiments.

MR. RUBIN: Yes. Correct. Correct. And AGR 7 and 8 are specifically designed for that validation.

MEMBER CORRADINI: Okay.

MR. RUBIN: Specifically designed, very clean validation experiment.

Another thing that you could see here is what is the fission product release or distribution rate in an intact particle. This is what MELCOR has predicted for an irradiation over two and a half years and what it represents is the build-up of cesium in the various particle constituents over time starting with the kernel on the left and then moving to the right through the various layers, and you can see that cesium -- the key barrier there is the silicon carbide layer.

What this also -- and you can see on the lower right what the actual releases of the particle over time, over that period of time are.

What is important here is that having the distribution for all the particles in the core as well as the matrix, for that matter, become the initial conditions for the accident.

So you have to be able to go through this to create a time-zero initial condition of fission product distribution in the core for the heat-up. So, this is a little example of having done that for a particle that's been irradiated for two and a half years.

In terms of validation and development and benchmarking, again you can see that the AGR program is going to provide opportunities for doing all that, for both MELCOR and PARFUME.

For the most part, where we are dealing with fission product release data, both PARFUME and MELCOR can utilize that irradiation, actual condition test data for that purpose. Where there is data specifically on particle failure rates it's really PARFUME that would best take advantage of that data.

However, even there, because reporting this response surface, or reporting this failure fraction curve into MELCOR, it will give us an opportunity to see if we're kind of like way off, conservative or not, given that it's particle -- so AGR-6, -7 and -8 will be a good validation test for our response surface curve that we're putting in there, and that will be the actual failures, and then we'll see what our response surface is predicting.

But it's simply empirical. It's a much more powerful validation test for the mechanistic codes.

Okay. The last thing that I would like to talk about is something that's already been completed. One of our objectives was to develop a set of critical attributes in the manufacture of fuel, high-quality fuel, high-performing fuel.

Those are listed here. The intent of having this was to provide kind of a workbook or a reference manual for an inspector to -- who would inspect the fabrication of TRISO particle fuel at the fuel fabrication plant, presumably the NGNP fuel fabrication plant.

So, this would give the inspector with some background a really good detailed understanding of what's important, many, many things are important, and then how to best construct their inspection plan.

That has been published now in May of 2009. It's a wonderful document. It gives you a good overview of how fuel is made, what is real important, how do you control those important things, et cetera, et cetera.

The basis for that was Oak Ridge's experience in developing the equipment, the procedures, the methods for making the AGR-1 fuel. They learned a lot. They understood what it took to make really good fuel and then we just asked them for a brain dump, if you will, on put it -- package it so others can use that information. And they did a very good job.

That's all I have.

DR. BASU: Any questions? If not, the

next presenter is going to be Joe Kelly, presenting the experiments supporting the thermal fluid model development effort.

(Off-record comments.)

MR. KELLY: At the subcommittee meeting I discussed the -- an overview of the evaluation model and in particular the PARCS-AGREE codes, as well as the experimental program and for this meeting I've been asked to just address the experimental programs.

And so I want to point at the screen. I'm still and old-time presenter. Excuse me.

So the point of this is that the majority of the experimental database is to be provided by the Department of Energy and the applicant. An example, that would be the fuel qualification program which Stu just talked about.

The NRC program supplements the DOE program, and the idea is to provide data for code validation and model development.

And, remember, when we put some of this n place we were targeting 2013 as having an initial version of the evaluation model with validation. So, we had to be very aggressive at starting this work.

The objective of the presentation is to give you a high-level overview of the NRC experimental program with an emphasis on the integral tests.

There are two integral tests and five separate effects tests. These are basically targeted to the area of thermal fluids. The first test that I'll talk about is the high-temperature test facility at Oregon State University, and then I'll talk about an OECD program to be conducted at the HTTR reactor.

It's a loss of force circulation test. And that would be at JAEA. That program is pending agreement but we plan to participate.

The separate effects test, there is a flow and heat transfer test in a Pebble bed at Texas A&M. Prismatic core heat transfer also at A&M. That program hasn't started yet, so that's why it's grayed-out and I won't be discussing it today.

Air ingress flow test at Penn State. Emissivity of vessel components, Wisconsin, and that program is complete and so, again, I will not be discussing it today.

And finally, bypass flow study in a prismatic core at Texas A&M.

So, this is the high-temperature test facility at Oregon State University. It's an integral effect test. It's a joint DOE and NRC program. The scale of the facility is one-quarter, both in height and diameter, so the vessel is approximately six meters tall and an outside diameter of two meters.

Reduced power, our max power is 2.2 megawatts. It is a full-temperature facility, so we are going to -- the inlet and outlet temperatures here are for the prototype plant, the MHTGR 350, and that's what it's being scaled to.

It's reduced pressure, so maximum pressure is eight bar. So, during the depressurization phase we will only catch the tail-end of the depressurization.

The initial core configuration is a prismatic block and that's scaled to the MHTGR-350 design of General Atomics. We have the option of going back in and putting in a Pebble bed core should we decide to go that way. That would be in a follow-on program.

The question was asked earlier about what was the status of the program and I will just briefly address that.

The Oregon State University has actually built a building for the facility. That's complete. All of the infrastructure work for the power supply has been done. The contract has been let for the construction of the vessel itself, as well as the balance-of-plant, and that's a steam generator and the circulator, and also a reactor cavity simulation tank.

CHAIRMAN ABDEL-KHALIK: Is this electric heating?

MR. KELLY: Pardon me?

CHAIRMAN ABDEL-KHALIK: Electric heating or nuclear heating?

MR. KELLY: Electric.

MEMBER POWERS: Electric, yes.

MR. KELLY: I'll show a little bit more about what it looks like inside in a minute. If you look at the top of the vessel you can see the inlet plenum with simulators for the control rod guide tubes.

And then, as you go down through the core you can see the blocks that will simulate the prismatic core. I'll show you more details on that in a second.

The orange block here is for the permanent side reflector. You can see the outlet plenum with the post. One of the objectives was to keep the actual details of the design as much like the prototype as possible.

This is the annular outlet, you know, inlet-outlet duct, the crossover duct to the steam generator, and the other one is provided for both vessel access and also later, possibly in a follow-on DOE program, visualization where they would try to make laser measurements in the outlet plenum.

MEMBER CORRADINI: So this is an APECS scaling lodge? You're going to catch the tail-end of the blow-down?

MR. KELLY: Yes. Fortunately, because it's not a light-water reactor the blow-down is not terribly important.

MEMBER CORRADINI: Sure. Yes, I understand.

MR. KELLY: Yes, and I'll --

MEMBER CORRADINI: But, all the other links scale -- all the scaling that I see here reminds me, is APECS, as far as I can tell.

MEMBER BANERJEE: Is it part of APECS?

MR. KELLY: No. No.

MEMBER CORRADINI: It's a separate building.

MEMBER BANERJEE: Separate facility?

MR. KELLY: Yes.

MEMBER BANERJEE: And at what stage is that?

MR. KELLY: The building, Oregon State

just built it and provided it to us. The tower supplies have been put in and we are now constructing the vessel and the balance-of-plant.

A contract has been let to a company in Portland, and they are doing that. It's the same company that built APECS.

MEMBER CORRADINI: Seaward?

MR. KELLY: Harris Thermal.

MEMBER CORRADINI: Okay. Yes.

VICE CHAIRMAN ARMIJO: How is the heating done, again?

MR. KELLY: I'll show you.

VICE CHAIRMAN ARMIJO: Okay.

MR. KELLY: But they are graphite electrodes. Okay?

VICE CHAIRMAN ARMIJO: Right.

MR. KELLY: The purpose of this is to provide data for the validation of thermal fluid analysis tools. And, in particular, the HTTF was designed to model the depressurized conduction cool-down transient.

That would be initiated by a double-ended guillotine break for the annular cross-over duct, but we'll also be looking at other, more probable types and locations of breaks, things like control rod drives and instrumentation breaks.

The facility will have a reactor cavity cooling system to provide the decay heat removal capability in a passive way, but also to provide a well-characterized boundary condition for the vessel.

The facility will also be higher pressure scenarios, like the pressurized conduction cool-down in normal operation, at least for prismatic only, by using nitrogen as a simulated coolant instead of helium.

So this is what a DCC, the bearing looks like. This is the primary purpose of the facility. It has four distinct phases. The initial phase is, of course, the depressurization, the flow --

MEMBER CORRADINI: Can we go back a minute?

MR. KELLY: Sure.

MEMBER CORRADINI: I'm sorry. So, you're one-quarter scale, which means the heat fluxes all have to go up by a factor of two if you're going to scale outside the -- if you're going to scale this? Is that approximately right?

MR. KELLY: Yes.

MEMBER CORRADINI: Okay.

MR. KELLY: Yes. The -- by volume, it's

1/64th --

MEMBER CORRADINI: Right.

MR. KELLY: But for natural circulation in the vessel, the power is 1/32nd.

MEMBER CORRADINI: So you're increasing all the heat fluxes in any one position by a factor of two?

MR. KELLY: Yes.

MEMBER CORRADINI: Okay.

MEMBER BANERJEE: And has here been a scaling analysis, top-down?

MR. KELLY: Yes. Yes. They followed a , you know, hierarchical two-tier approach, and there is a draft scaling report and the final report will come out once the facility has been built so the as-built conditions will go into it.

So this is the transient that has been designed for. It starts out with the depressurization which only, depending upon the size of the break may only be a very few seconds.

Certainly, for a large break, as is pictured here -- the second part of this, you have the reactor vessel full of high-temperature, very low-density helium. The reactor cavity is an air, air-helium mixture that is relatively cold and much denser.

So, at a horizontal break, such as is pictured, you can get a stratified countercurrent flow. It's also called lock exchanged where, even though it's two gases, they actually act more like, you know, what we would be thinking of as water and steam.

So you can set up a countercurrent flow where the outlet plenum can rapidly flood with the -- again, higher-density air-helium mixture. Once the outlet plenum is full, you go to a phase which might be diffusion-dominated where you have to get the air up into the top of the vessel before you have enough of a buoyancy driving force to start natural circulation, which is what's shown here.

And so it's important to get the phases right or pretty much right. Not only does the natural circulation give you the air -- the rate of air ingress into the facility, but the timing is important.

You want to know when this starts because that has an impact on where and how much graphite oxidation occurs.

VICE CHAIRMAN ARMIJO: But you have to get all that helium out of the bottom vessel, don't you, before outlet starts?

MR. KELLY: You mean in here?

VICE CHAIRMAN ARMIJO: Yes. Yes.

MR. KELLY: That's really a stagnant space.

VICE CHAIRMAN ARMIJO: It is?

MR. KELLY: Yes. So this is the outlet plenum here. Let's see, if I go back --

MEMBER CORRADINI: But the main intent -- I just want to make sure that I understand the global intent of this. The global intent of this is not to make it appropriately-scaled to catch the accident, but it's to make it appropriately-scaled enough to test the calculations?

MR. KELLY: Exactly.

MEMBER CORRADINI: Okay.

MR. KELLY: That's right.

MEMBER CORRADINI: All right. Because --

MR. KELLY: You want to make sure you are in the right ranges of parameters --

MEMBER CORRADINI: Right.

MR. KELLY: -- things like Reynolds numbers so that --

MEMBER CORRADINI: Right.

MR. KELLY: -- you're testing a code in

the right place.

MEMBER CORRADINI: I think, to answer Sanjoy's point, they've done a scaling analysis. The scaling analysis is, as Joe has indicated, but because you're going down to quarter scale you are distorting a number of things.

So, they are trying to minimize distortion strictly for the long -- strictly for the goal of computational comparison versus empirical one-to-one comparison, right? These are not going to get empirical one-to-one.

MEMBER BANERJEE: Yes. The concerns I have, of course, are many with scale --

MEMBER CORRADINI: You always have concerns.

MEMBER BANERJEE: Yes. One of them, of course, is, to the extent that these fluids are turbulent and the countercurrent behavior is determined by the density gradients and the effects on the turbulent scales, actually, the vertical scales across that interface which determine all the mixing phenomena.

You know, it's different with a gas -- liquid vapor flow. At least you can mock that up in some way. Here, I think the effective scale in phenomena like that is more difficult to evaluate, and that's why I asked you about the scaling because there's a microscale which determines mixing phenomena which is dependent on the macroscale and how the energy cascades down, the eddies chain, so --

MR. KELLY: Yes. And we will not match the Reynolds numbers.

MEMBER BANERJEE: Right.

MR. KELLY: Of the inlet flow because, like you say, the link scale.

MEMBER BANERJEE: Yes.

MR. KELLY: But at least -- we will be turbulent, you know.

MEMBER BANERJEE: Yes. You'll be turbulent --

MR. KELLY: So it's not --

MEMBER BANERJEE: But then what you have to do, and this is very, very difficult in any CFD codes, is to get the effective scale on the turbulence right, especially for these densities stratified interfaces.

MR. KELLY: Yes.

MEMBER BANERJEE: This is a huge problem.

MR. KELLY: Yes. And from the NRC perspective, we're not using CFD as part of our evaluation model. We have a more or less empirical model that has been built in the MELCOR to handle that countercurrent flow, and so this will be a test of that.

MEMBER BANERJEE: Right. But the problem then, is what is the effective scale. I mean, how do you determine that this -- you can fit it to this data.

MR. KELLY: Yes.

MEMBER BANERJEE: But then how do we know it works on a larger scale?

MR. KELLY: Well, we do have some smaller-scale experiments that we can at least compare between those, but we won't have anything larger than this. Quarter scale is pretty large.

MEMBER BANERJEE: There's a huge amount of data in the atmospheric dispersion, particular of heavy gases. There have been a lot of tests done on mixing at stratified interfaces on pretty large scales, so you might look at that data, actually.

MR. KELLY: Okay. Thank you.

You know, to answer the previous question about the space in the bottom of the vessel, this is the outlet plenum here and there are posts that support the core.

Now, in reality, what sits underneath here -- you can't see it in this picture -- is the shutdown cooling system and it's designed so that that area is stagnant during normal operation and only if that circulator comes on do you then bring flow down into this.

Of course, there is some small flow paths for pressure equalization and so on, but in general this is a stagnant area. It will eventually fill up with the air.

CHAIRMAN ABDEL-KHALIK: If you go to slide number six.

MR. KELLY: Yes.

CHAIRMAN ABDEL-KHALIK: And look at the picture on the right and think about it in the long term -

MR. KELLY: This one --

CHAIRMAN ABDEL-KHALIK: Right. You have a break at both the inlet point and the exit point of this natural circulation loop are essentially the same pressure.

MR. KELLY: Yes.

CHAIRMAN ABDEL-KHALIK: So what is the pressure at the top in the upper plenum at the top of the circulation loop?

MR. KELLY: Well, in this case, this is the reactor cavity.

CHAIRMAN ABDEL-KHALIK: I fully understand what's going on.

MR. KELLY: Which is only slightly above atmospheric.

CHAIRMAN ABDEL-KHALIK: But would you agree that the inlet and exit points in this picture for this natural circulation loop are at the same pressure?

MR. KELLY: Do you mean right here?

CHAIRMAN ABDEL-KHALIK: Right.

MR. KELLY: Within a pascal or two, yes.

CHAIRMAN ABDEL-KHALIK: Right.

MR. KELLY: Yes.

CHAIRMAN ABDEL-KHALIK: So, what is the pressure in the upper plenum of this vessel relative to the building pressure?

MR. KELLY: You know, you're only talking fractions of an atmosphere.

CHAIRMAN ABDEL-KHALIK: I'm just trying to figure out whether or not you would ever have natural circulation in this scenario.

MEMBER BLEY: What's the thermal driving --

CHAIRMAN ABDEL-KHALIK: It would be pretty tiny.

MEMBER BLEY: Yes.

MR. KELLY: Well, it's a very -- it is a small -- depending upon what your loss coefficients are, it's a small flow, but the center of the core here is like at 1600 degrees C. Okay.

The outside here is about 400 degrees C, so you've got a huge thermal driving head.

Now, you're correct that if this were full of helium the natural circulation is almost zero. It's only if it's full of a much heavier gas, like air, that you will have natural circulation.

CHAIRMAN ABDEL-KHALIK: So you're saying this works only because you have radiation heat transfer out of the outside surface of the vessel -

MR. KELLY: Yes. That's the way it works.

CHAIRMAN ABDEL-KHALIK: So that the plenum is a little cooler?

MR. KELLY: That's the way the plant works. During a depressurized conduction cool-down or --

MEMBER BLEY: The purpose arrows are slightly cooler than the red arrows. But the annular region is just enough cooler to make -- boy, that's pretty delicate, isn't it.

MEMBER BANERJEE: No, but it was. I mean, I'm sure it --

MR. KELLY: Well, I mean, from a design standpoint, if you can prevent this natural circulation, you're happy, because this natural circulation is not cooling the plant down. The plant is -- the plant is -- basically it's radial conduction.

CHAIRMAN ABDEL-KHALIK: Just follow one single streamline from the inlet all the way to the exit, and tell me what the pressure is at the top of this upper plenum.

MR. KELLY: I've done a calculation, but I don't remember the values but, I mean, you're talking a hundred pascals or something. It's not much, but you are talking about a 1200-degree C difference in the gas mixture temperature, so that does -- the d rho dt times 1200 degrees gives you something, but that's not all there is. It's not a high flow rate.

But that -- I think there's a misconception here in how the plant works, and that may not have ever been explained to you. So what -- the way it works -- let me actually go back --

MEMBER CORRADINI: I think -- I think what Joe's trying to say is it would be lovely if there were no flow.

MR. KELLY: Yes.

MEMBER BLEY: But he's going to tell us why now.

MR. KELLY: Well, you don't want the flow because it brings oxygen in and causes oxidation, et cetera. And that's one of the reasons -- now, I'm not presuming to design here, but if I were, the steam generator in the actual design is below the vessel.

One of the reasons is you don't want natural circulation through the vessel. So, how do you cool it? The idea is it's passive.

What's not shown here is the reactor cavity cooling system which surrounds this and so those are panels that are either air-cooled or water-cooled, depending up the design. That's what eventually takes the decay heat out to the roof and releases it to the environment.

So what you're relying on is the heat transfer from the vessel wall to the reactor cavity cooling system, and that's primarily thermal radiation.

The vessel walls at peak temperature would be about 450 degrees C during a DCC event. And so it's about 80 percent thermal radiation and 20 percent natural convection. You know, there will be a natural convection loop between the hot vessel an these relatively cold panels.

But that's how the heat gets out, but how it gets to the vessel wall is the combined conduction and radiation paths. You conduct the heat radially outward through the core to the core barrel, then there is a gap between the core barrel and the vessel which is primarily radiation heat transfer and then, again, radiation to the RCCS.

That's where, if we had more time and we could take you through the plant design it would have, you know, helped some.

MEMBER BLEY: Thank you.

MEMBER POWERS: We could come back to schematic you've set it up.

MR. KELLY: This one?

MEMBER POWERS: Yes. It's highly-idealized.

MR. KELLY: Very much so.

MEMBER POWERS: And through reality of the plant is, there's a certain -- going to be some sort of leakage out the top. Is that enough --

MR. KELLY: You know, in the facility we will do other different breaks, like a control rod drive break, and that would allow the heavier gases coming straight down, and it would be a different transient.

MEMBER POWERS: So let me focus just on this one.

MR. KELLY: Okay.

MEMBER POWERS: How much leakage from peripheral parts of the plant changes this drawing?

MR. KELLY: I don't -- I could only speculate.

MEMBER POWERS: And look at that in your experimental program.

MR. KELLY: Well, you know, most of the leakage paths are relatively small, especially if we are talking about a --

MEMBER POWERS: That's why I asked you how much of a leakage does it take to upset this picture. If I broke the top off, as I know, that upsets it, but there must be at some point where some small leakage -- and there's a bit of a problem you always have in helium systems is you always have a certain amount of leakage.

MR. KELLY: Yes. I'll be able to --

MEMBER POWERS: How much leakage does it take to upset this picture?

MR. KELLY: I'll be able to give you a better answer later. Once we have our computation tools in better shape we can do sensitivity analyses.

MEMBER POWERS: And that's a specific part of your computational plan?

MR. KELLY: Yes.

MEMBER POWERS: Can we see that matrix?

MR. KELLY: Not yet.

So, this is what the core is going to look like. There will be ten of these plates. One for each fuel element level in the actual core. The plate is made of a designer ceramic to model the moderator graphite, and we had to do that because we needed to scale -

MEMBER POWERS: What kind of ceramic is --

MR. KELLY: We design it. We design it to a specific thermal conductivity.

MEMBER POWERS: What materials, zirconia, yttria, alumina?

MR. KELLY: I don't know. I'm sorry.

MEMBER POWERS: But it set the thermal conductivity of graphite?

MR. KELLY: Pardon me?

MEMBER POWERS: That's your intent, to have the thermal conductivity of graphite?

MR. KELLY: Actually, one-eighth the thermal conductivity of graphite.

MEMBER POWERS: One-eight.

MR. KELLY: Of irradiated graphite. And the reason is --

MEMBER POWERS: Wow, that is a designer --

MR. KELLY: This is quarter scale.

MEMBER POWERS: Oh, okay.

MR. KELLY: In the radial direction, but we want the same temperature drop across the core during a DCC, when all of the decay heat in the core is conducted radially outward, so we needed to scale the thermal conductivities of the facility to get that temperature drop so we can, you know, see if there are -- to what degree natural circulation occurs.

So, these are the core plates. On the outside of these would be the permanent side reflector. In the core we have these repeated hexagonal arrays. We have one of these for each fuel element stack, so there will be 66 of them for the MHGGR design.

There are graphite electrodes at the center and at the vertices. The coolant channel holes are the same diameter as in the prototype and give us the same porosity as the prototype.

In the plate it -- the center part of this models the inner reflector. You'll notice there are six holes in it. Those are for the control rods -- where the control rod guide tubes are and they are flow paths and we use them both for the -- to model the coolant flows for the control rods, as well as for the bypass for the gaps that we're not modeling.

Likewise, there are holes around the periphery for in the side, what would be the side reflector, and those also are to model the flow through the control rods, as well as -- for control rod cooling as well as the bypass gas.

MEMBER POWERS: I can see how you would design a ceramic to have the contribution to the thermal conductivity to be just about any number you wanted.

I don't know how you get the photon contribution through the thermal conductivity scale.

MR. KELLY: And I can't address that. I -- I do know -- what they are doing is, they will make a batch, mold it and send it out for testing and characterize the thermal conductivity of those samples.

MEMBER POWERS: If I send to a commercial outfit and say, "Tell me the thermal conductivity," and I'm very elaborate, they will give me that thermal conductivity up to a thousand degrees. That's not good enough for you.

MR. KELLY: Right. They -- you know, that's part of the spec is that it has to go up to 1600, so they're measuring the thermal conductivity as a function of temperature over that range.

MEMBER POWERS: Yes, but if it's just a batch they will only measure the phonon contribution. Without the whole you don't get the photon contribution.

MR. KELLY: Right.

MEMBER POWERS: I mean, I assume it's -- probably similar to carbide --

MR. KELLY: No, but we are going to measure the emissivity a well, if that's where you're headed.

MEMBER POWERS: Yes. How are you going to scale that?

MR. KELLY: You don't.

MEMBER POWERS: You're stuck with whatever you get?

MR. KELLY: Pretty much. So actually,

earlier, when Professor Corradini was talking about the heat flux, one of the problems with the reduced-scale facility, you know, typically would be heat loss, but in this course our heat loss is the passive decay heat removal, that's one of the things we want to model.

So we want the vessel wall to be at the same temperature as the prototype. But how do you do that? We actually can't affect the natural circulation heat transfer loop very much.

That -- it is what it is and you can't do much about it. But for the radiation there is a slight difference and it has to do with what we've chosen for materials.

In the actual plant it will be something like SA-508 and we're going to use -- you know, 508, and the emissivity of that is much closer to Point A, you know, once it's been at temperature for a while.

Whereas, we're using a stainless steel, 304 stainless, and the emissivity of that in an air environment at 500 degrees C is about .35, so that helps us scale the heat loss.

And likewise, our RCCS panels, instead of deliberately making them be a high emissivity, we're choosing power stainless steel to keep the emissivity low.

So the next program is actually a test reactor. It's an OECD program at the high-temperature engineering test reactor which is run by the JAEA in Japan, and it's -- be for loss-of-forced-circulation.

So, this is a real graphite-moderated, helium-cooled reactor. The total power is 30 megawatts.

The proposed test program which, you know, the agreement is pending, should be signed hopefully by the end of this month. It would be a series of three loss-of-forced-circulation tests.

The table shows the initial conditions for those three runs. Run number two is the one at full power, which is 30 megawatts, and you see it's also at full temperature. The other two cases are parametrics about that at lower power and lower temperature.

Now, for all cases in the test it is complete loss-of-forced-circulation, so all the gas circulators are tripped. There's no forced flow in the primary system at all.

Also, the reactor is not scrammed, so these are ATWS. Run number three, which is a parametric study about this, they also trip their vessel cooling system pumps. So, in effect, what that does is, it makes the reactor cavity cooling system not active, and that's why run number three has to be a low-power test.

So this is the -- shows the expected behavior. What we have is a plot of reactor power versus time, so log, scale and time. You'll notice it's focused in so it's at a lower power.

The blue curve is for a case where the initial power was at 30 megawatts or full power. The red curve at nine. And what you see is, the reactor is predicted to shut itself down, and that's affect -- you know, as it heats up, because of the negative -- strong negative temperature feedback, but also because of xenon.

As the xenon decays away, the reactor will come back to power. It will go re-critical again and reach a new steady state power at a lower -- lower power level.

So this will provide a very good test for coupled, you know, thermal fluids reactor physics and we'll be modeling this with PARCS-AGREE and also with MELCOR, but MELCOR only up to the point of re-criticality, because there's only a point kinetics model in MELCOR.

MEMBER BLEY: Joe, I want to interrupt you for just a second.

MR. KELLY: Yes.

MEMBER BLEY: You're about a third of the way through your slides. We have another presentation on graphite, primarily because we thought some members who weren't at the subcommittee would want a chance to hear about it or at least ask questions about graphite.

I kind of need to recalibrate what we do here. Maybe if we save ten minutes for the graphite discussion, and if somehow you can pick and choose and try to close in about ten minutes.

MR. KELLY: I think --

MEMBER BLEY: There's a lot of stuff here that everybody wants to get into, but we just don't have time.

MR. KELLY: I think the rest of it will go a little bit faster because they are all on separate effects tests and I'm not showing very much, other than just telling you what they are.

MEMBER BLEY: Okay. If you can summarize that way I think that will be good.

MR. KELLY: Because of time.

MEMBER BLEY: Yes.

MR. KELLY: So, this was at the Pebble bed flow and heat transfer test at Texas A&M as part of a cooperative -- actually, part of the same cooperative agreement as the Oregon State one and they're looking in four different areas for Pebble bed pressure drop, the radial porosity distribution and they came up with a very unique way of measuring that.

Those tasks are finished. They are now looking at the radial velocity profile and the idea of this is to try to get some quantification of what's known as the wall bypass effect.

That's where the Pebble bed meets the reflector interface and then finally, pebble-to-gas convective heat transfer. And so what we're doing is inductively heating one pebble within this randomly-packed bed of pebbles.

And in moving that pebble all the way from the center of the core out to right up against the reflector, so that we can get -- characterize the heat transfer coefficient both as a function of Reynolds number and the position.

This shows the air ingress flow test at Penn State University, and we talked about this stratified countercurrent flow again.

The picture on the right is a water brine test, so the density difference isn't -- isn't nearly as large as you get between air and helium. The scoping studies with water brine are complete. We'll be doing scaled experiments with helium air. Those have now started.

So we look at break orientation, the L over D effect and also break geometry. In the picture here, on the right-hand side, this would be the reactor cavity. It's water brine, so it's a little bit heavier than the water that would be in the reactor vessel.

This is the break plane. So initially there's a very large flow rate of the brine coming along underneath the pure water with the pure water coming up into here.

The intermediate stage is when this level reaches the bottom of the crossover duct. At that point the flow rate decreases significantly. The final stage is when the mixture level hits the top of that and the flow rate almost stops.

And my last slide is the prismatic core bypass flow study. This is just starting at Texas A&M and, as you know, bypass is very important to quantify in both prismatic and Pebble beds. This is for prismatic.

So, we'll be using a matched index of

refraction, coupled with particle image velocimetry in order to measure the flow, both in the bypass gaps and the coolant channels. There are also will be pressure-drop measurements for the bypass gap in coolant channel.

And the point of this test is to give us the -- an experimental basis for the loss coefficients that we need in order to model these bypass flows, just to give us model development information for a more macroscale computer.

The test section is comprised of three prismatic columns. They are stacked two blocks high so, by moving the columns apart with different shrouds we can change the thickness of the bypass gaps and we can also separate the columns axially to change the thickness of the gap at the horizontal interfaces, as well as we can put wedge-shaped gaps in, because that's more what we expect to see in the actual plan due to the irradiation damage.

And that's all. Are there any other questions?

(No response.)

DR. BASU: So, I guess the last presentation is going to be by Dr. Srinivasan on the graphite performance.

DR. SRINIVASAN: Good morning. My name is Makuteswara Srinivasan. I work at the Office of Nuclear Regulated Research, specializing in the nuclear graphite.

Excuse me while I find out how to --

For the lack of time, what I'm going to do is to briefly go over in about five minutes the entire presentation because we don't have time.

I do have prepared remarks which I will give so that you will have the transcript for the entire presentation.

The most important thing, salient point, is that we started the graphite research back in 2002 with Exelon, potentially coming for a design certification application.

We abandoned that right after Exelon abandoned their plans, but then, since 2005, the Energy Policy Act, we institute our research program in 2006.

Basically, the original research plan that was written back in 2002 was still valid and it was taken up by worldwide research organizations and they were working on, to that plan, more or less.

As a result of that, even by 2006 the majority -- some of the salient points that we addressed were being addressed by the other research programs around the world.

Subsequent to 2006 a lot of programs that were mentioned in the 2003 plan are -- they're being addressed by the DOE program itself. So, we have subsequently modified our program and condensed it and basically we only try to do confirmatory research.

The staff is getting input by participating in worldwide and national -- worldwide meetings such as the Nuclear Graphite Specialist Meeting. We also participate in the IAEA graphite knowledge-base development program.

We participate in the generation for the international forum graphic working group committee meeting, as well as another program with OECD, NEA, with respect to international graphite program on irradiation creep.

The staff has also participated in development of codes and standards continuously from 2003 -- actually it was NRC who initiated the program with ASME. We also initiated a program with ASTM to have nuclear graphite specification.

As a result of our initiatings, ASME currently has a Division 5, Section 3 graphite core case which is a draft case that will be published by the end of this calendar year.

Once again, by -- as a result of our initiating we have an ASTM specification for graphite components that are actually two specifications; one for high-irradiation material in the graphite core; and the other, for graphite support components which do not see that much radiation, nevertheless, play an important role such as graphite core supports.

Through all these efforts NRC has been pushing the international community to think not only in data-gathering, but to understand what the data means in terms of the uncertainty in data, model uncertainty, the data uncertainty, the uncertainties that could be involved in inspection, as well as in the structural integrity analysis.

We continue to participate and, once again, inform that our main thrust will be to understand and appreciate the uncertainties involved so that when we do establish margins, the same margins for minimum requirements or properties are discussed in temperature limits, we have sufficient margin commensurate with the uncertainties and unknown information.

Moving on, there are currently two programs that NRC is addressing in the graphite area.

One is with respect to the stored energy release. That is primarily because graphite, when it is irradiated, accumulates irradiation damage and stores energy.

Subsequently when -- if graphite is heated to temperatures more than the temperature for which it was exposed, then there is potential for release of this stored energy in the form of heat.

The prevailing theory is that this stored energy release is a concern for graphite -- for reactors operating at temperatures about 300 degrees Celsius.

The consensus is that, because the NGNP HTGR in high-temperature gas reactors are going to be operated at temperatures greater than 300 degrees Celsius, this would not be a concern.

However, there are no experiments and data to confirm that, and that's why we took this research initiative and this is being performed at Oak Ridge National Laboratory.

And basically they have samples of graphite that have been irradiated in the temperature regime of the NGNP area. As well, because of the irradiation programs being conducted at INL, they will have samples that are more or less representative of the NGNP HTGR conditions.

So, the objective of this research will be to take these samples and to conduct a systematic stored-energy release experiment so that we'll have some information.

The second program is with Argonne National Laboratory and that has to do with -- to develop an independent confirmatory analysis tool for NRC, to ensure that the design -- the applicant, when they give us the design data will have sufficient margins in their stress analysis of graphite components.

Basically the tool that we are developing at Argonne National Laboratory will be elaborate and it will have spatial stress distribution, and it will consider the important properties that contribute to deformation such as the thermal expansion, the shrinkage and expansion characteristics, the thermal conductivity was on refuel differences models -- in models differences, as well as heat flux and the temperature distribution that occurred in the core. We expect to have this tool by the end of 2013.

There are other graphite programs that could be of importance and that has to do with, for example, the perennial question about the rapid oxidation of graphite or loss during potential accidents or moisture ingress and so forth.

The position that the staff is taking is that it is the applicant's responsibility to give us all the data as well as the rationale, technical basis for developing their design.

The staff will participate, once again, in international meetings and keep aware and abreast of what is happening in those areas.

I believe that concludes my presentation.

MEMBER BLEY: Srini -

DR. SRINIVASAN: Yes.

MEMBER BLEY: At the subcommittee meeting, Tom Kress asked a couple of questions. You just sort of answered one of them, but I want to put them back on the table for this meeting.

DR. SRINIVASAN: Okay.

MEMBER BLEY: One he mentioned, he didn't detect any plans in the research to evaluate the potential for steam graphite reactions in the event of water ingress, and you've just put that off on the applicants.

But where do you folks stand on that? You're not going to do any research? Do you think that's well-understood? Where do --

DR. SRINIVASAN: To the best of my knowledge, basically that is included in the INL program, and so within the constraints of what we have in the NRC DOE memorandum of understanding, we will be participating in the setting up of experiments and, you know, actually monitoring what's going on and analyzing the data.

When the data become available we will make independent judgment and arrive at our conclusions. That probably -- yes, it is not --

MEMBER BLEY: It's not under your research plan, but it's covered -- you'll be looking at it through the memorandum of understanding?

DR. SRINIVASAN: Right. In a way, in our research plan what we have done is that whatever research that we are not actively pursuing, we will be actively participating in international meetings and gathering information and such insights.

So, the staff will become aware of what's going on. And, not only that, the staff will also provide our input in the experimental stage itself with respect to the number of samples that a statistical distribution, what kind of an uncertainly might propagate and those kinds of issues.

So, we may not be conducting the

experiments, but we fully intend to be aware of what's going on, and incorporate whatever the outcome of those research into our thinking and the design assessment.

MEMBER BLEY: Okay. The other one he raised is about the -- he indicated -- well, and others did, too, that to transport efficient products around a primary circuit is likely to be dominated by the graphite dust problem for contributions to a source term, and he said he knows of no models for dust production under high irradiation conditions and temperature, nor for dust aerosol characteristics, and that this would be a large source of uncertainty in source terms, and that's -- I don't think that's in the research plan, either.

DR. SRINIVASAN: No. Originally, we had tribology, the understanding the source of how the dust is created during reactor operations, but -- you know, we excluded that, but then we have, for example, tribolic aspects in graphite fuel matrix abrasion and dust generation.

It is there, once again, as an active participant wherever the information is going. At this point in time the only thing that I am aware of is the work being conducted at Chinua University, both in the -- from the fundamental point of view of tribology of the wear and friction studies as well as having some experimental model of a tube and creating dust and the seeing how the deposition and dust deposition goes.

But these are in the preliminary stages, and I have been talking with Professor You on that just to understand what -- how things are.

MEMBER BLEY: What do we know from experience with graphite reactors about dust production?

DR. SRINIVASAN: The dust production apparently in the prismatic reactor is almost virtually not there. In the AGR, British AGR, as well as Magnox reactors -- Magnox reactors, dust production is virtually no.

In the case of AVR, yes, there was dust production, and the concern is that we have to look at the -- apparently have to look at the dust -- amount of dust that is created versus the volume of the entire enclave, if you will, and considered.

But, I am not -- I am not cognizant or I am not aware of all the fission product issues and things. In terms of dust generation, dust generation is possible more in the Pebble bed reactor than in the prismatic reactor.

MEMBER BLEY: Well, if they go toward a Pebble bed, you're going to have to do something and --

DR. SRINIVASAN: Yes. We have issues with respect to the -- in the Pebble bed case, there is -- there is a possible -- the dust generation between the Pebble and the Pebble, which is a different kind of a thing because the A3 matrix is a different material than the graphite, than between the Pebble and the graphite core component itself, really.

So, those are the -- but we have to understand those issues and we will remediate that.

MEMBER POWERS: Well, I think we know a little more -- I mean, we know something about dust, and we do know that there are some work going on at the University of Missouri to try to characterize dust coming from two mechanisms.

One is the abrasion mechanism and the other is what happens when you have a gas particle conversion. We also know that a lot of the dust formation that occurred in AVR occurred because of injection of contaminated materials that subsequently will paralyze, and that, too, is going to afflict these reactors because every time you open up and refuel you will introduce inevitably things that will paralyze, and whether or not you get gas particle conversion every time you introduce helium because helium always carries a certain amount of oxidants with it.

So, we know some things. Now, are there mechanistic models for dust production? I'm not aware of -- of any right now, because I think, as Srini says, our understanding of the particle production is probably a little limited right now.

DR. SRINIVASAN: One thing I might add also is that there are no standardized test method to understand. And, in fact, in about four weeks I'll be making a plea to ASTM organization to initiate a standardization test exclusive for NGNP kind of a thing of a pebble-to-pebble, as well as pebble-to-solid, heightened rich helium environment, because there are issues with respect to conducting experiments in conditions other than what is expected under NGNP/HTTR environment.

MEMBER POWERS: One thing that maybe you could help me understand a little better. I see persistent studies of graphite at various times in the Russian literature, and it seems to persist most in the Siberian Russian literature.

But I'm told by those in the business of manufacturing graphite that the graphites being investigated in Russia are inferior to the graphites that will be considered for these reactors and so I shouldn't pay too much attention to that.

What is the situation?

DR. SRINIVASAN: I'm not sure whether I understood your question, but are you talking about the purity levels of something or -- I'm not --

MEMBER POWERS: Well, I see, every time I look, which is not too often, I see publications -- very interesting work going on in Russia on graphite.

DR. SRINIVASAN: Okay.

MEMBER POWERS: And when I mention it to those in the business of producing graphite I am told in no uncertain terms that the Russian graphites are inferior to the kinds of graphites that they would propose here and, therefore, I should not pay any attention to those.

And I do, but -- but I don't know whether -- whether I should or not.

DR. SRINIVASAN: You are right. I think that has been the perennial problem with the graphite industry and that's one of the reasons we have now ASTM's standards, specification exclusively about nuclear graphite.

And it has -- that -- there are two specific -- there are two specifications, as I said, one for graphite exposed to high temperatures and high doses of irradiation, and another one that is exposed to low doses of irradiation.

Both of them are very strict in terms of impurity content in terms of metallic materials and so forth. And the nuclear graphite basically undergoes -- undergoes some halogenation in their manufacture so that the heavy metals are complete -- you know, removed much -- you know, there are specific values that we have to conform to.

So, in terms of purity levels -- oh, and the purity specification also considers effects in decommissioning, you know, exposure and so forth, too. So, if the Russian materials have to come to the United States or elsewhere and if they have to have graphite specifications, then they have to conform to ASTM specification.

Additionally, the ASME core also invokes ASME specification, and therefore, there are strict protocols and the INL folks, in their program, are instituting much more stringent matters in qualification.

I might also -- I should also add that the ASTM specification in the last -- about the quality control issues invokes that the manufacturer has to conform to NQA-1 qualification requirements.

So, hopefully, the -- oh, the other thing that I should say in passing is that the Idaho National Laboratories have completed their baseline characterization program. And from what we know from the baseline characterization, the graphites that they study have much, much better properties than even -- you know, the ones that we used in Fort St. Vrain and Peach Bottom reactors.

Now, the challenge on the graphite manufacturer is to reproduce these things on the same strictness and so forth, and that's why ASTM specification is there.

I don't know whether I answered you fully, but that's the status that I know.

MEMBER POWERS: Well, you gave me some useful insight. I will admit that -- if I could indulge just another question, you have a program, or there is a program to look at the change in properties as you irradiate graphite.

The problem the analyst comes about is that when he goes through in a thermal excursion, you will start to anneal those properties, and you get more rapid annealing along basal planes than you do axial planes, or the other way around. I can't remember which it is.

And he -- he has a material that's polycrystalline. How does he average?

DR. SRINIVASAN: Good question. The ASTM specification for nuclear graphite is really isotropic or neoisotropic grade. So basically we have taken out -- based on the British experience and so forth, the British folks did contribute to the development of ASTM specification.

Basically the ASTM specification calls for isotrophy in proportion to thermal expansion of less than 1.05, up to that. So, any kind of a -- you know, the graphite that do not meet the isotropic criterion will not be in the reactor as a core material.

VICE CHAIRMAN ARMIJO: But wouldn't you have to -- you'd have to fabricate it so you don't get an anisotropic properties.

MEMBER POWERS: But you're doomed to get anisotropic behavior as you go through the thermal anneal because --

VICE CHAIRMAN ARMIJO: Not if the crystals are totally random. You know, any one -- any one crystal is doomed --

MEMBER POWERS: That's the question I'm asking is: How do I assure that I have no anisotrophy when I know that I am annealing across basal point faster than I am in the axial?

VICE CHAIRMAN ARMIJO: Right. But, you know, you've got a big, huge composite -- the same thing with zirconium alloys, right, you don't want this preferred orientation where all the crystals are lined in certain ways, then you start getting really anisotropic properties in the bulk.

But within a single little crystal, you're absolutely right, but if you have millions of crystals oriented --

DR. SRINIVASAN: Now, let me -- let me address that as follows, really. One of the programs, for example, that initially we had in 2003 was a concern that, even for isotropic materials, as Dana says, if you are not careful in things for big bodies that you are making for a code material, there could be variation.

And therefore, Idaho National Lab, the program basically had the material taking a different -- you know, they had a very strict mapping, if you will, of the samples that are being tested, different orientations at different locations and so forth.

So, we're -- going in, we know the baseline characterization at both -- you know, whatever the -- the grain, and against the orientation and in between. So far, the information is that it is conforming to the isotropy requirements.

Now, one thing that has not been done as yet is what Dana -- Dr. Powers is asking, and that is, after conducting the version of thermal expansion studies as a function of temperature, we have not annealed and then pre-done the testing in this -- to the best of my knowledge in INL.

However, the work was done in pattern on another material, and basically with isotropic material the information remained the same as -- to the best of my knowledge.

MEMBER POWERS: Yes. The problem is, you have to do the partial anneal. You know, anneal some temperature and time, then stop and look at it and then anneal farther and stop and look at it, because that's what the analyst has to analyze, in that transient, and the way the transients were going in this because there's so much heat capacity in the block, the relatively slow transients, so he's spending a lot of time in that intermediate phase.

And at least the calculations I've seen, they -- people assume no heat release during annealing and they assume consistent isotropy -- in isotropy -- consistent isotropy.

And when I ask why they -- why they can do that, they said, well, they don't have any choice because they don't have any data to the contrary and it's easier to do the calculations, all of which I concede. But I just don't know what the truth is.

DR. SRINIVASAN: I think part of your -- part of your question could be, hopefully, answered in our Oak Ridge National program for the stored energy program.

And the other one is that we all -- the NRC staff has always the opportunity to -- any new shifts coming up in things, it would make the applicant's responsibility to answer that. If it is something that needs to be of a confirmatory nature then, of course, you know, we will try to get resources to conduct those.

MEMBER POWERS: Yes. One thing that does happen to you if you irradiate long enough, you get non-annealable damage, which is typically the precipitation in basal planes in the graphite, and you anneal amount that will never anneal, and that will introduce anisotropy into the system.

But I don't know whether you'd go long enough to do that or --

DR. SRINIVASAN: One other thing I wanted to mention is that in the ASME core development, we also had a lot of discussions about some of the uncertainties in this kind of knowledge that is not there, and so forth, and I've been successful so far, at least, to push for adequate conservatism in against stressing the stress and temperature limits, incorporating the uncertainties in data and model and also even when you do as many experiments in the irradiation -- irradiated examples, it could still be not adequate for statistical purposes and so forth.

MEMBER POWERS: Well, thank you --

MEMBER CORRADINI: I'm looking at the Chairman who is going to tell me to shut up, right, but I had one --

MEMBER BLEY: I'm about to.

MEMBER CORRADINI: -- connected question with Dr. Powers. Does the staff view this as a -- does the staff view this as a safety issue that, as I start with the NGNP in one state and as time marches on, I have radiation damage and thermal effects that will be dimensional changes, and those dimensional changes will cause a change in flow which will cause essentially a change in the hot spot.

Is that a safety issue or is that an operational issue that DOE is in charge of giving you assurances and calculations for?

DR. SRINIVASAN: It is a -- it is a safety issue and the deformation is especially, you know, is a safety issue because you have to have -- the control rods and fuel rod panels have to maintain their integrity so that the safety is maintained.

So, the Argonne National Laboratory program that we are doing is one of the information is to confirm the design, applicant's design deformation limits.

MEMBER CORRADINI: Okay.

DR. SRINIVASAN: So, that is an important issue.

MEMBER CORRADINI: And this is -- this is an Argonne program?

DR. SRINIVASAN: Argonne National Laboratory--

MEMBER CORRADINI: Okay.

DR. SRINIVASAN: -- we have a program.

I think you also asked another question. I forgot.

MEMBER CORRADINI: No. You've helped me out.

MEMBER BLEY: I'd like to thank Srini very much for your presentation, sir, and invite all the committee members to our next subcommittee, since this is such a topic of conversation.

Mr. Chairman.

CHAIRMAN ABDEL-KHALIK: Thank you very much.

At this time we are scheduled to take a break for 15 minutes and when we return we will be off the record.

(Whereupon, the meeting concluded at 10:42 a.m.)

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High Temperature Gas-Cooled Reactor (HTGR) NRC Research Plan
An Overview

Sudhamay Basu, RES
(sudhamay.basu@nrc.gov)

Advisory Committee on Reactor Safeguards
May 13, 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 and request a letter from ACRS

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

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 Areas
  - Probabilistic risk assessment (PRA)
  - Human factors (HF)
  - Instrumentation and control (I&C) technology

Assumptions

- Research scope in large part generic
- Availability of data from DOE-sponsored VHTR R&D
- Availability of applicant-furnished data for licensing requirements
- Availability of complementary data from international HTGR R&D programs
- Reliance on national and international codes and standards
- Adequate resource allocation

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 University of 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 (starting soon)
  - Core heat transfer and bypass flow studies at TAMU
  - Stratified counter-current flow experiments at PSU
  - Emissivity experiments at University of Wisconsin

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Implementation Status

Fuel Performance and Fission Products

- Fuel fission products (FP) model development
  - Modeling FP diffusion through coatings and matrix
  - MELCOR FP code development and benchmarking
  - PARFUME code exercise and benchmarking
- Fuel failure modeling in MELCOR and PARFUME
- Coordination with DOE/INL on AGR 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

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Implementation Status
Structural Analysis and Process Heat

- Structural Analysis
  - Assessment of concrete behavior at high temperature
  - Seismic and Soil-structure interaction of deeply embedded structure
  - Seismic loading consideration for multi-modular design
- Process Heat Utilization
  - Assessment of incident blast loading on reactor
  - Thermal-fluid behavior of process heat components
  - Component degradation issues
  - Toxic and corrosive gas dispersion modeling
  - Tritium migration modeling

Implementation Status
Digital I&C, Human Factors, and PRA

- Instrumentation and Control (I&C)
  - Research initiated on advanced reactor controls and instrumentation
  - Investigate advanced diagnostics and prognostics (AD&P) system integration issues
- Human Factors (HF)
  - Developing technical basis to support update of HF guidance documents
- Probabilistic Risk Assessment (PRA)
  - Planning study undertaken to identify PRA needs and scope for HTGR licensing
  - Other HTGR- related PRA activities

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

Abbreviations

Abbreviation	Stands For
ACRS	Advisory Committee on Reactor Safeguards
AD&P	Advanced Diagnostics and Prognostics
AGC	Advanced Graphite Creep
AGR	Advanced Gas Reactor
AGREE	Advanced Gas-cooled REactor Evaluation
ANL	Argonne National Laboratory
ANS	American Nuclear Society
DOE	U.S. Department of Energy
FP	Fission Products
HF	Human Factors
HTGR	High Temperature Gas -Cooled Reactor
HTTR	High Temperature Test Reactor
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratory
I&C	Instrumentation and Control
LOFC	Loss of Forced Circulation

Abbreviation	Stands For
MELCOR	System code for modeling severe accident phenomena
NGNP	Next Generation Nuclear Plant
ORNL	Oak Ridge National Laboratory
OSU	Oregon State University
PARCS	Purdue Advanced Reactor Core Simulator
PARFUME	PARticle Fuel ModEl
PRA	Probabilistic Risk Assessment
PSU	Pennsylvania State University
R&D	Research and Development
RPV	Reactor Pressure Vessel
SCALE	Standardized Computer Analysis for Licensing Evaluation
SG	Steam Generator
SNL	Sandia National Laboratories
TAMU	Texas A&M University
VHTR	Very High Temperature Reactor
V&V	Validation and Verification

Thank You

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HTGR Fuels
Research and Development

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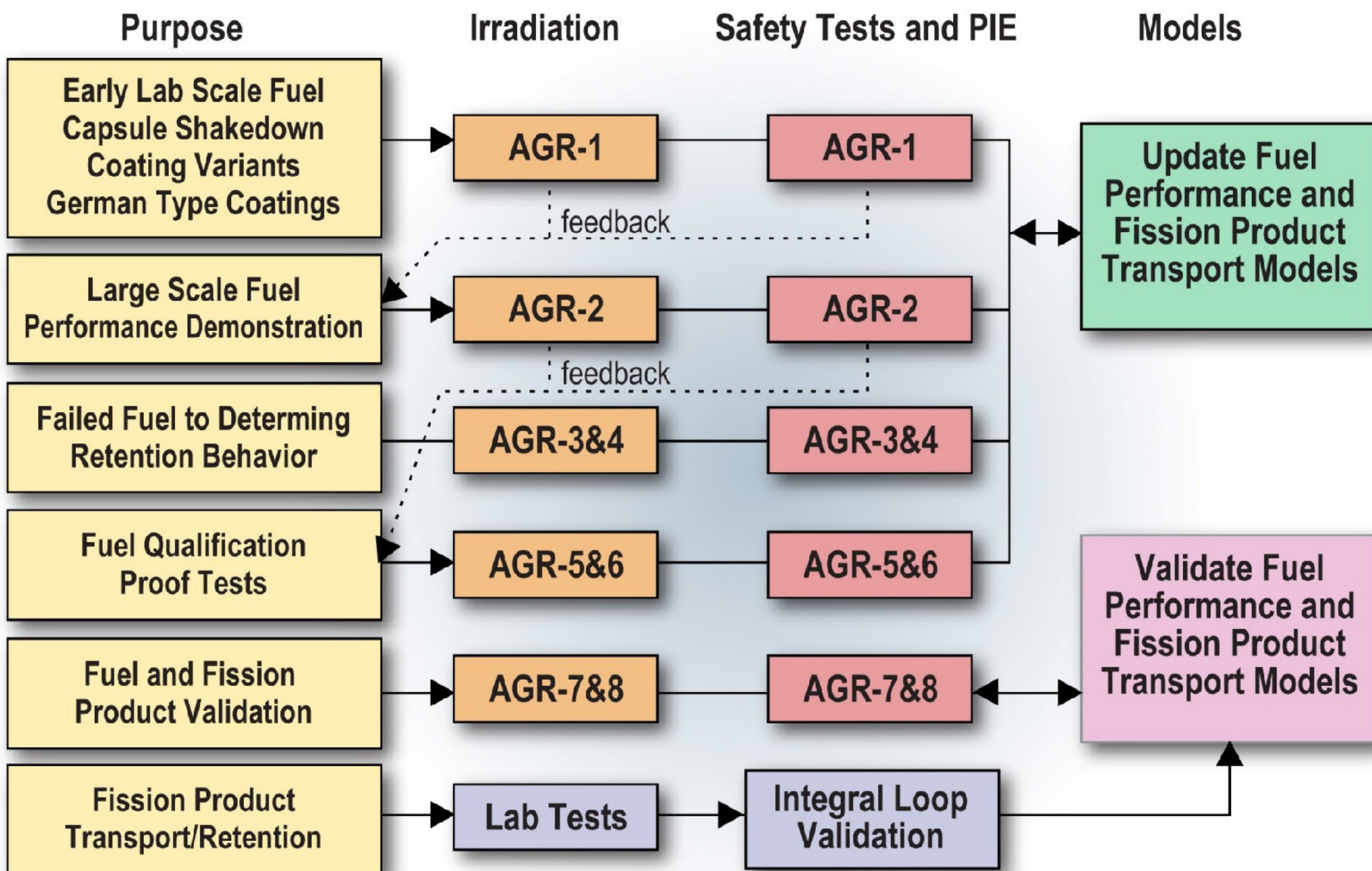
Advisory Committee on Reactor Safeguards

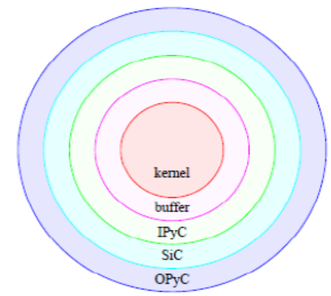
May 13, 2011

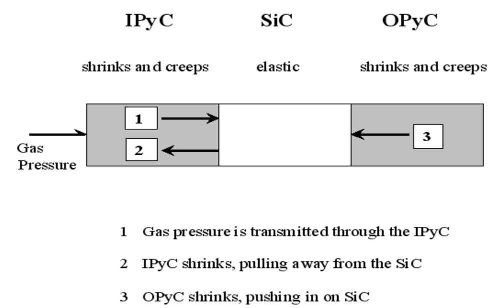
NGNP/AGR Fuel Program Activities

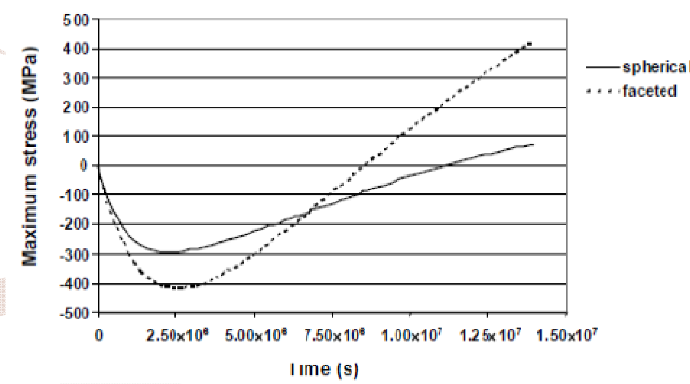


PARFUME

- A state-of-the-art HTGR TRISO coated fuel particle (CFP) performance code developed by INL
- Simulates CFP thermal, mechanical and physico-chemical behavior
- Models internal overpressure, layer cracking, de-bonding and asphericity failure mechanisms
- Accounts for statistical variations in CFP geometric and material properties
- Calculates failure probability of a population of CFPs
- Calculates metallic and gas fission product transport in intact CFPs, failed CFPs, fuel matrix, graphite element
- Used to support DOE's AGR fuel design, development and qualification activities
- NRC/DOE MOU allows NRC to acquire PARFUME to support NRC's review of the NGNP license application







PARFUME Modeling*

- **Fuel element thermal analysis**: $\rho c_p \frac{\partial T}{\partial t} = k \bullet \nabla^2 T + \dot{q}$
  - Cylindrical fuel compact in prismatic block
  - Spherical fuel pebble
- **Fuel particle thermal analysis**: $\cancelto{0}{\rho c_p \frac{\partial T}{\partial t}} = k \bullet \nabla^2 T + \dot{q}$
- **Fuel particle stress-strain analysis:** $\frac{\partial \varepsilon_r}{\partial t} = \frac{1}{E}\left(\frac{\partial \sigma_r}{\partial t} - 2\mu \frac{\partial \sigma_t}{\partial t}\right) + c(\sigma_r - 2\nu\sigma_t) + S_r + \alpha_r \dot{T}$
- **Fuel particle failure probability analysis:** $P_f = 1 - e^{-\int_V \left(\frac{\sigma}{\sigma_0}\right)^m dV}$
- **Fuel particle FP transport analysis**: $\frac{\partial C}{\partial t} = -\nabla \bullet J + S \quad J = -D\left(\nabla C + \frac{Q' C}{RT^2} \nabla T\right)$
  - Intact particles, failed particles
- **Fuel element FP transport analysis**: $\frac{\partial C}{\partial t} = -\nabla \bullet J + S \quad J = -D\left(\nabla C + \frac{Q' C}{RT^2} \nabla T\right)$
  - Intact particles, failed particles,
  - Uranium contamination
- **Fuel element FP release analysis**

* MELCOR will have similar modeling - except for bullets in **red font;**
(i.e., MELCOR will use/input failure fraction curve or response surface)

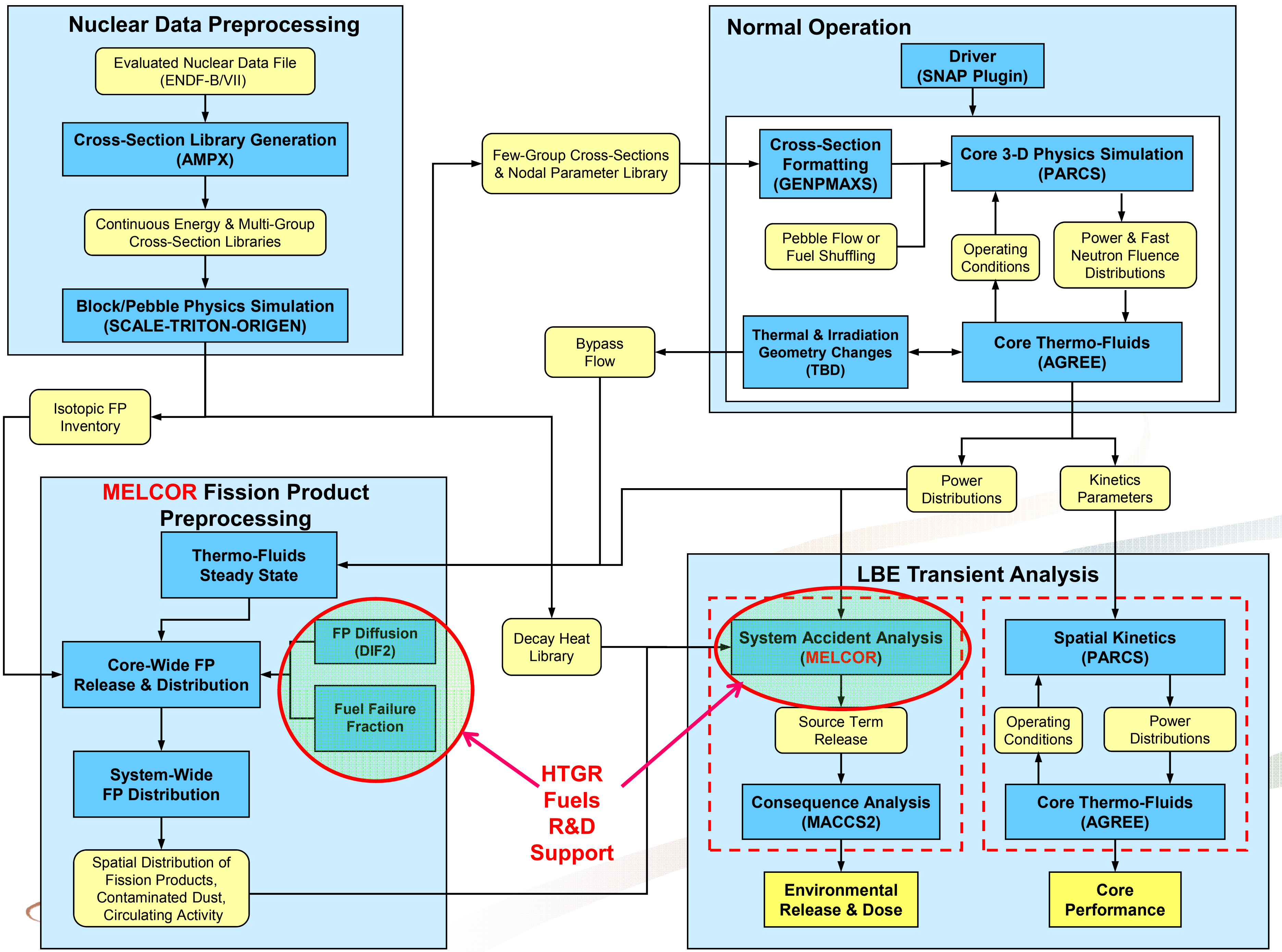
PARFUME Future Development Activities

Targeted PARFUME Code Parameters	INL PARFUME Development Activity*	Planned AGR Data Source	Data Availability (CY)	Targeted Completion (CY)
PyC (e.g., irradiation dimensional changes, creep, strength, elastic modulus); SiC (strength, creep, elastic modulus, thermal expansion); kernel (CO production, swelling)	Update particle mechanical, physical properties	In-pile and out -of -pile experiments	2011-2015	2015
Benchmark fission gas and fission metal releases	Pre-test and post -test predictions of fuel irradiation and accident tests	AGR-1	2011-2013	2013
Benchmarks fission gas and fission metal releases	Pre-test and post -test predictions of fuel irradiation and accident tests	AGR-2	2011-2016	2016
Update fission gas and fission metal diffusion coefficients	Pre-test and post -test predictions of fuel irradiation and accident tests	AGR-3&4	2012-2017	2017
Benchmark particle failure rate models; Benchmark fission gas and fission metal releases	Pre-test and post -test predictions of fuel irradiation and accident tests	AGR-5&6	2013 -2018	2018
Validate particle failure rate models; Validate fission gas and fission metal release models	Pre-test and post -test predictions of fuel irradiation and accident tests	AGR -7&8	2015 -2020	2020

* NRC to conduct PARFUME development in parallel with INL

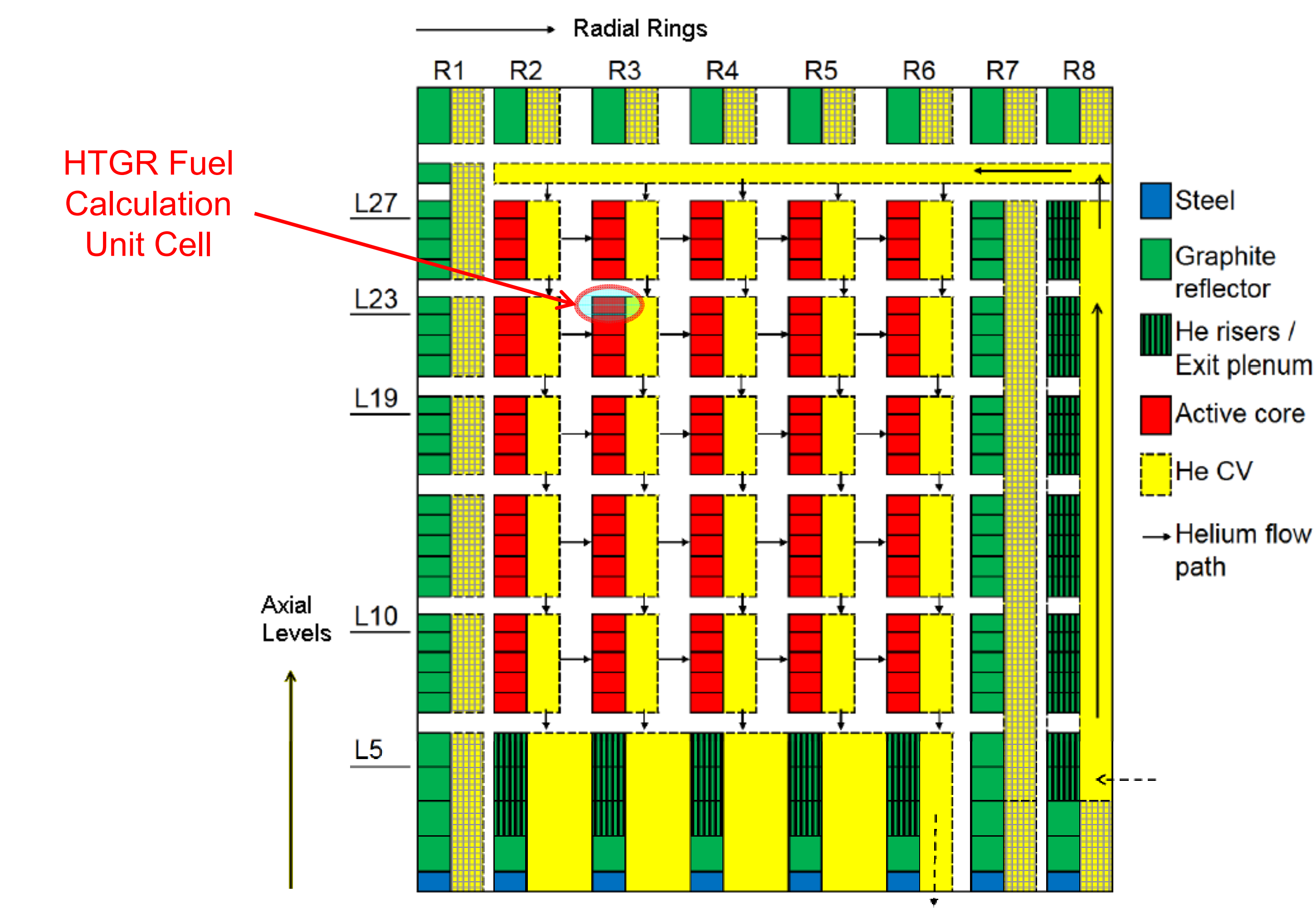
Planned Uses of PARFUME

- Assess the fuel particle failure probability model used in the NGNP applicant's safety analysis evaluation model
- Assess the effects of variations in fuel particle manufacture, operating conditions and accident conditions on NGNP CFP failure probabilities
- Support MELCOR core-wide fission product model development and verification via code-to-code benchmark predictions
- Provide knowledge and training to NRC technical staff on fuel particle behavior, failure and fission product transport
- Assess the effects of NGNP startup testing and safety testing on NGNP fuel operational performance
- Provide insights on the effects of potential fuel-related compensatory measures (e.g., limits on fuel temperature, burn-up) on the NGNP source term and licensing



Nuclear Data Preprocessing
Evaluated Nuclear Data File (ENDF-B/VII)
Cross-Section Library Generation (AMPX)
Continuous Energy & Multi-Group Cross-Section Libraries
Block/Pebble Physics Simulation (SCALE-TRITON-ORIGEN)
Isotopic FP Inventory
Few-Group Cross-Sections & Nodal Parameter Library
Normal Operation
Driver (SNAP Plugin)
Cross-Section Formatting (GENPMAXS)
Core 3-D Physics Simulation (PARCS)
Pebble Flow or Fuel Shuffling
Operating Conditions
Power & Fast Neutron Fluence Distributions
Thermal & Irradiation Geometry Changes (TBD)
Core Thermo-Fluids (AGREE)
Bypass Flow
Power Distributions
Kinetics Parameters
MELCOR Fission Product Preprocessing
Thermo-Fluids Steady State
FP Diffusion (DIF2)
Fuel Failure Fraction
Core-Wide FP Release & Distribution
System-Wide FP Distribution
Spatial Distribution of Fission Products, Contaminated Dust, Circulating Activity
Decay Heat Library
HTGR Fuels R&D Support
LBE Transient Analysis
System Accident Analysis (MELCOR)
Source Term Release
Consequence Analysis (MACCS2)
Environmental Release & Dose
Spatial Kinetics (PARCS)
Operating Conditions
Power Distributions
Core Thermo-Fluids (AGREE)
Core Performance

MELCOR Core Model



MELCOR Fuel FP Transport Modeling (Unit Cell Phenomena/Mechanisms)

Prismatic Fuel Elements

- Generation distribution in fuel kernel
- Recoil in kernel to buffer layer
- Diffusion through kernel
- Diffusion through layer coatings (buffer, IPyC, SiC, OPyC)*
- Diffusion through compact matrix
- Generation in compact matrix (failed, intact particles and U contamination)
- Vapor transport across fuel gap (fuel compact-to-graphite block web)
- Diffusion through graphite block web
- Vaporization transport at graphite block web-to-coolant channel
- Radioactive decay during diffusive transport

Pebble Fuel Elements

- Generation distribution in fuel kernel
- Recoil in kernel to buffer layer
- Diffusion through kernel
- Diffusion through layer coatings (buffer, IPyC, SiC, OPyC)*
- Diffusion through pebble matrix
- Generation in pebble matrix fueled region (failed, intact particles and U contamination)
- Vaporization transport at pebble matrix surface-to-coolant interface
- Radioactive decay during diffusive transport

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* N/A for failed particles

MELCOR vs. IAEA CRP-6 Benchmarks

Case	1a	1b	2a	2b	3a	3b
	Cs-137 Fractional Release					
MELCOR	**0.465**	**1.0**	**0.026**	**0.995**	**1.00E-4**	**0.208**
PARFUME	**0.467**	**1.0**	**0.026**	**0.996**	**1.32E-4**	**0.208**
France	0.472	1.0	0.028	0.995	6.59E-5	0.207
Korea	0.473	1.0	0.029	0.995	4.72E-4	0.210
Germany	0.456	1.0	0.026	0.991	1.15E-3	0.218

(1a): Bare kernel (1200° C for 200 hrs)
(1b): Bare kernel (1600° C for 200 hrs)
(2a): Kernel + buffer + iPyC (1200° C for 200 hrs)
(2b): Kernel + buffer + iPyC (1600° C for 200 hrs)
(3a): Intact particle (1600° C for 200 hrs)
(3b): Intact particle (1800° C for 200 hrs)

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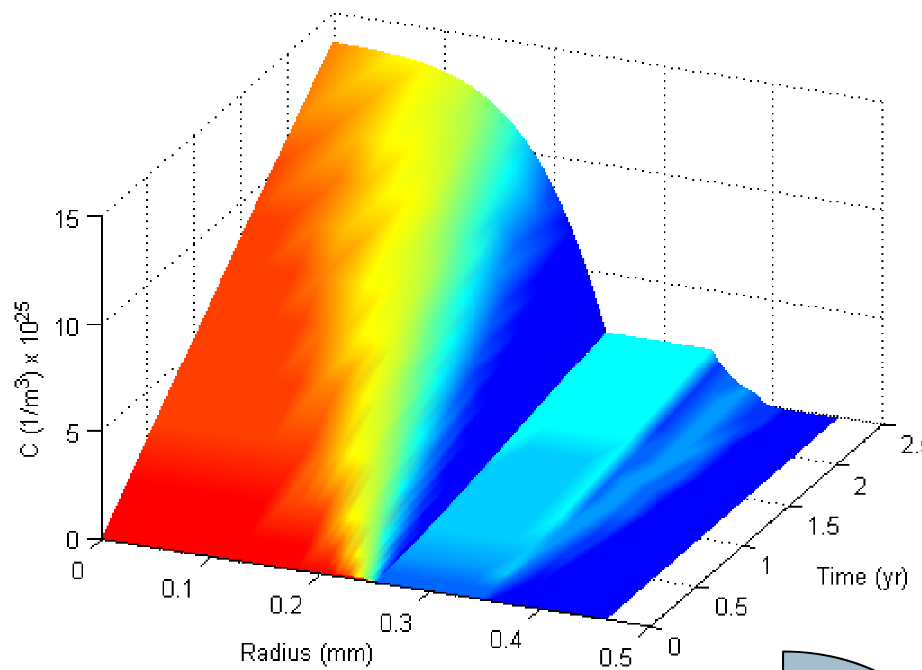
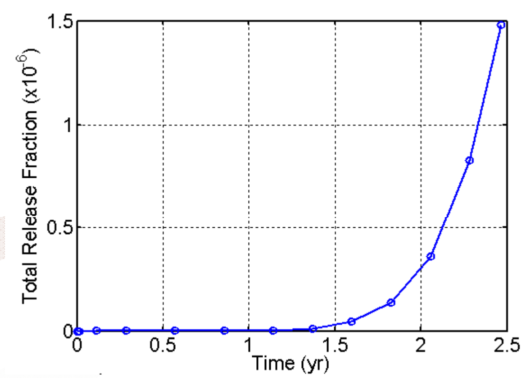
MELCOR CFP FP Distribution & Release

MELCOR Diffusion Solution

- Use core cell component temperatures (temperature dependent diffusion coefficients)
- Finite difference solver (DIF2) integrated into MELCOR
- Track intact and failed particles
- Output of the diffusion calculation is spatial distribution in the particles (kernel/buffer), graphite, and relative amounts released to the primary system (for each isotope from each core cell)
- FP distribution and release rates are scaled using ORIGEN results for burn-up (more accurate in terms of actual isotope inventory)

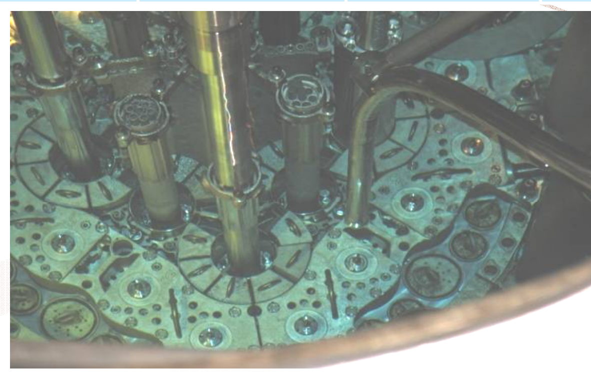




AGR Data for NRC Fuel FP Transport Modeling

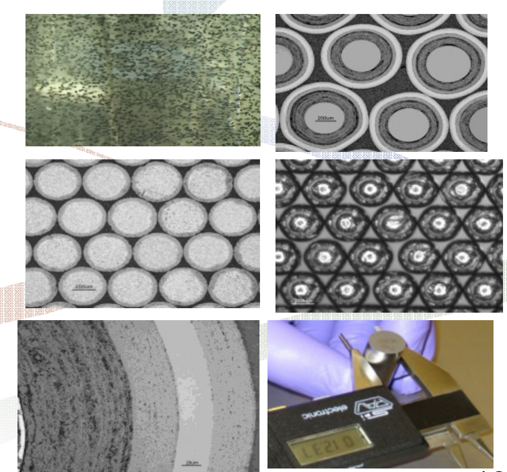
AGR	Purpose	Irradiation Data Use		Accident Condition Data Use	
		CFP Performance	Fuel FP Release	CFP Performance	Fuel FP Release
1	Irradiate lab-scale coater fuel; establish German coating on UCO design; investigate coating variants; shakedown irradiation capsule design	PARFUME	PARFUME MELCOR	PARFUME	PARFUME MELCOR
2	Irradiate large-scale coater fuel; demonstrate irradiation & accident condition fuel performance	PARFUME	PARFUME MELCOR	PARFUME	PARFUME MELCOR
3&4	Irradiate and accident condition test *large-scale* fuel to develop fission product transport data for NGNP fuel fission product transport models	PARFUME	PARFUME MELCOR	PARFUME	PARFUME MELCOR
5&6	Irradiate and accident condition test fuel to demonstrate (i.e., qualify) fuel performance during NGNP normal operation, design and beyond design conditions	PARFUME	PARFUME MELCOR	PARFUME MELCOR*	PARFUME MELCOR*
7&8	Irradiate and accident condition test fuel to develop V&V data for NGNP fuel performance and fission product transport models	PARFUME	PARFUME MELCOR	PARFUME MELCOR	PARFUME MELCOR
Lab Tests	Conduct tests to develop data for Modeling FP Transport in HPB and VLPC	N/A	MELCOR	N/A	MELCOR

*Air and moisture ingress effects data developed from AGR 5/6 PIEs



Fuel Fabrication Facility Oversight and Inspection

- Objective: Document the critical attributes for fabricating high quality, high performance HTGR fuel and guidance for developing a plan for inspecting HTGR fuel fabrication facilities
- Covers the following aspects:
  - Critical product parameters for fuel quality and performance
  - Fuel product inspection and testing equipment and procedures
  - Critical process equipment and process parameters
  - Calibration testing equipment and calibration inspection procedures
  - Maintenance procedures for fuel fabrication process equipment
  - Sampling methods, statistical analysis methods and acceptance criteria
  - Training and qualification of fuel fabrication facility staff
  - Fuel fabrication facility inspection guidance
- ORNL/TM-2009/041, Overview of Key issues and Guidelines for Regulatory Oversight and Inspection of HTGR Fuel Fabrication and Quality Control Activities, May 2009



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Abbreviations

Abbreviation	Stands For
Ag	silver
AGR	advanced gas reactor
CFP	coated fuel particle
CRP	coordinated research project
Cs	cesium
CV	control volume
DOE	US Department of Energy
FP	fission product
He	helium
HPB	helium pressure boundary
HTGR	high-temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory

Abbreviation	Stands For
IPyC	inner pyrolytic carbon
MELCOR	system code for modeling severe accident and containment phenomena
NGNP	Next Generation Nuclear Plant
OPyC	outer pyrolytic carbon
ORIGEN	Oak Ridge Isotope GENeration
ORNL	Oak Ridge National Laboratory
PARFUME	PARticle FUel ModEl
PIE	post-irradiation examination
PMR	prismatic modular reactor
SiC	silicon carbide
TRISO	tri-structural-isotropic
U	uranium
VLPC	vented low pressure confinement
V&V	verification and validation

Back Up Slides

MELCOR Fuel FP Diffusion Model

Particles
- Intact
- Failed
- Missing SiC
- U contamination

↓

Matrix

↓

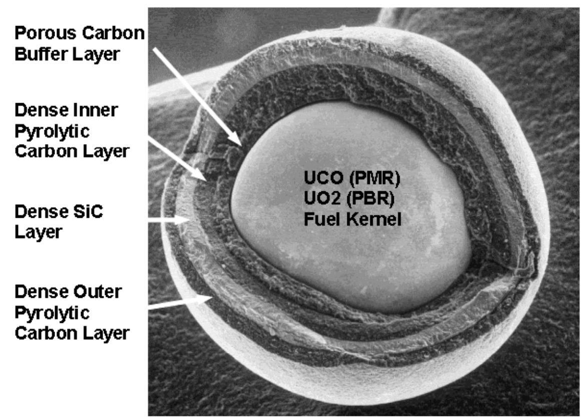
Compact/ Graphite Gap (PMR)

↓

Graphite (PMR)

↓

Coolant



$$\frac{\partial C}{\partial t} = \frac{1}{r^m}\frac{\partial}{\partial r}\left(r^m D \frac{\partial C}{\partial r}\right) - \lambda C + S$$

$$m = 1 \ (cylindrical)$$
$$m = 2 \ (spherical)$$

C = Concentration (kmol/m^3)
λ = Decay constant (1/s)
S = Source term (kmol/m^3-s)
D = Effective Diffusion coefficient (m^2/s)

$$D(T) = D_o e^{-Q/RT}$$

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Cs Transport in Silicon Carbide

- Goal: Increase the understanding of Cs transport in chemical vapor deposited SiC to asses the applicability of historical Cs diffusion data to new fuel fabrication
- Research Organization: University of Wisconsin – Todd Allen, Principal Investigator
- Approach: multi-scale modeling with experimental data
- Scope: investigate volume/bulk and grain boundary diffusion mechanisms (i.e., entire energy landscape)
- Modeling:
  - Inputs: SiC lattice structure, Cs formation energies, Cs migration energies
  - Calculated outputs: Cs concentration profiles and diffusion coefficients
- Experiments:
  - Characterize microstructure, distribution of grain boundaries
  - Measure concentration profiles of Cs in diffusion couple

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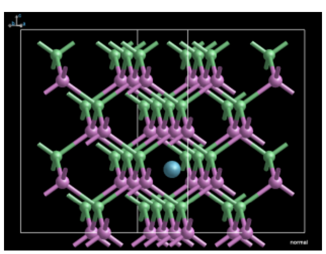
Cs Diffusion Modeling Results

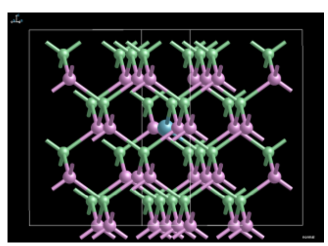
Bulk/Volume Diffusion

1. Interstitials too unstable to play a role in diffusion
2. Cs_c–$2V_{Si}$ defect structure is the stable form of Cs in bulk SiC
3. Cs_c–$2V_{Si}$ ring mechanism dominates bulk Cs diffusion in SiC—agrees with bulk integral activation energy
4. Cs defects too unstable to allow adequate Cs solubility— need additional mechanisms for Cs to enter SiC

1

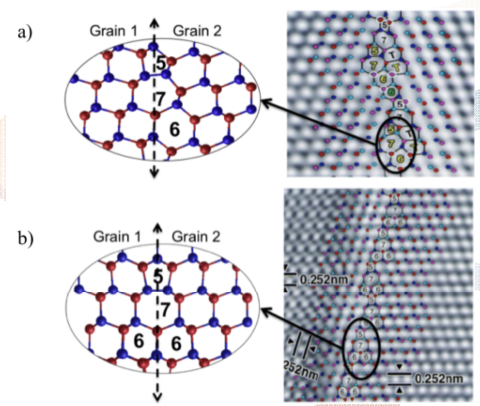


2,3



Grain Boundary Diffusion

- Strong tendency for Cs to segregate to grain boundaries
  - Even interstitial position is relatively stable
- Possible fast paths were found to be slow, even those that are fast for Ag
- No fast diffusion paths found



<u>New R&D</u>: Investigate possible effects of neutron irradiation on Cs diffusion in SiC

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Experimental Support for the NGNP Evaluation Model

J.M. Kelly
USNRC Office of Nuclear Regulatory Research
Joseph.Kelly@nrc.gov

Advisory Committee on Reactor Safeguards
May 13, 2011

Background

- **Supporting Experimental Program**
  - Majority of experimental database to be provided by DOE and applicant
    - (e.g.) Fuel qualification program
  - NRC program supplements DOE program
    - Provide data for code validation and model development
  - Presentation Objective
    - High-level overview of entire NRC experimental program
      - Emphasis on integral experiments

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Supporting Experiments

- **Integral Effects Tests**
  - High Temperature Test Facility (OSU)
  - OECD HTTR-LOFC Program (JAEA - pending)
- **Separate Effects Tests**
  - Pebble-Bed Flow & Heat Transfer (TAMU)
  - Prismatic Core Heat Transfer (TAMU)
  - Air Ingress Flow Tests (PSU)
  - Emissivity of Vessel Components (UW)
  - Prismatic Core Bypass Flow Study (TAMU)

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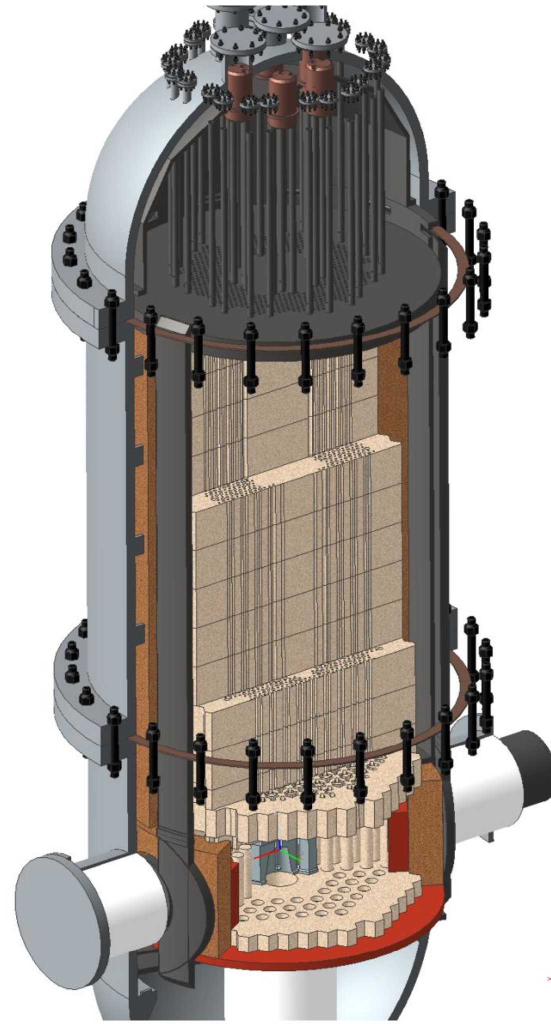
High Temperature Test Facility
Oregon State University

- **Integral Effects Test**
  - Joint DOE & NRC Program
  - Facility Scaling:
    - Draft scaling report complete, final report will use as-built information
    - ¼ length scale: 6.1 m tall
    - ¼ diameter scale: 1.92 m vessel OD
    - Power: 2.2 MW
    - Full Temperature
      - T_{in} = 259 °C
      - T_{out} = 687 °C
    - Reduced pressure: 8 bar max.
  - Core Configuration:
    - Initial: Prismatic Block (MHTGR)
    - Optional: Pebble-Bed



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HTTF Experimental Objectives

- **Provide data for validation of thermo-fluid analysis tools**
  - HTTF was designed to model the depressurized conduction cooldown (DCC) transient
    - Initiated by a double-ended guillotine break of the annular cross-over duct
      - Other break locations and sizes will also be examined:
        - (e.g.), CRDM & instrumentation line breaks
    - Reactor Cavity Cooling System (RCCS) to provide passive decay heat removal capability
  - Also applicable to other high-pressure scenarios by using N_2 instead of He as coolant
    - Pressurized Conduction Cooldown (PCC)
    - Normal operation (PMR only)

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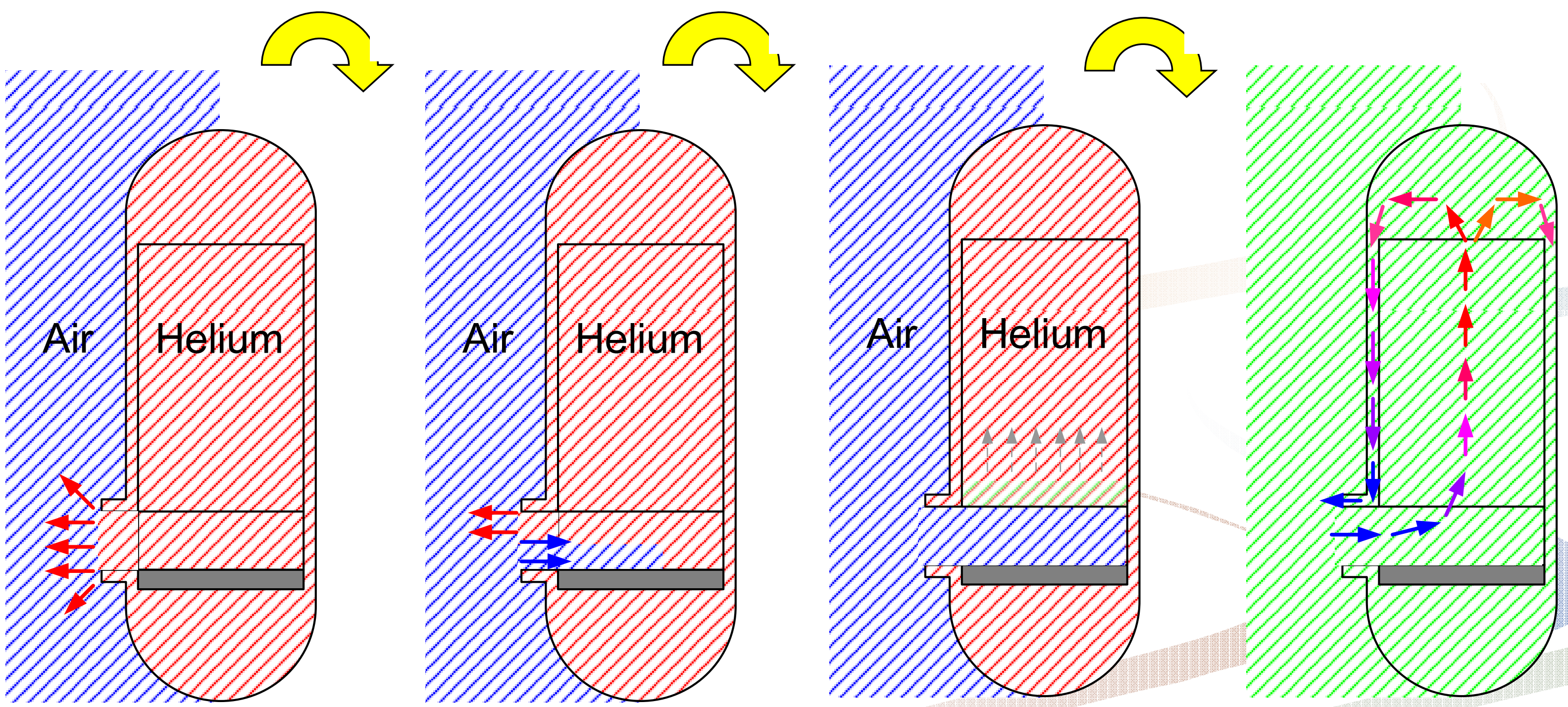
HTTF Experimental Objectives

- **DCC transient with air ingress**
  - Four distinct phases:



(a) Depressurization (b) Stratified Flow (c) Diffusion (d) Natural Convection

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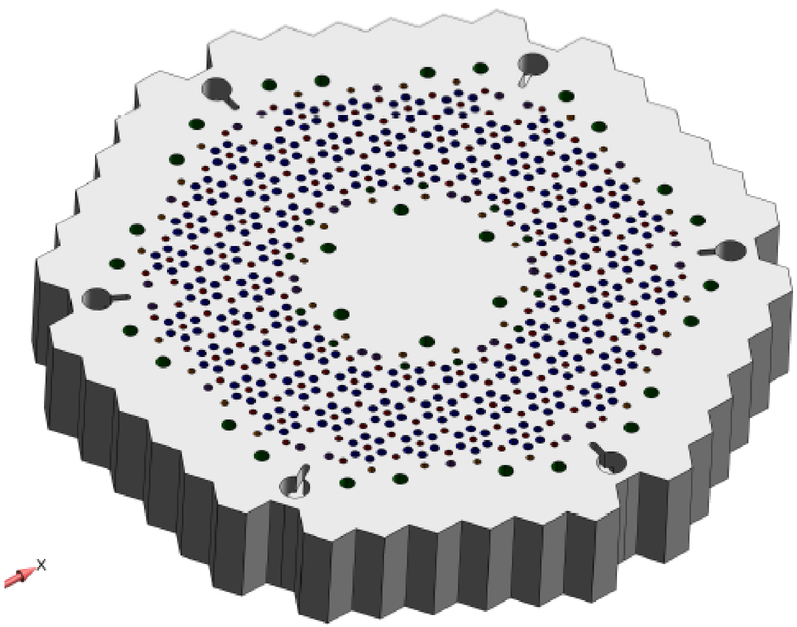
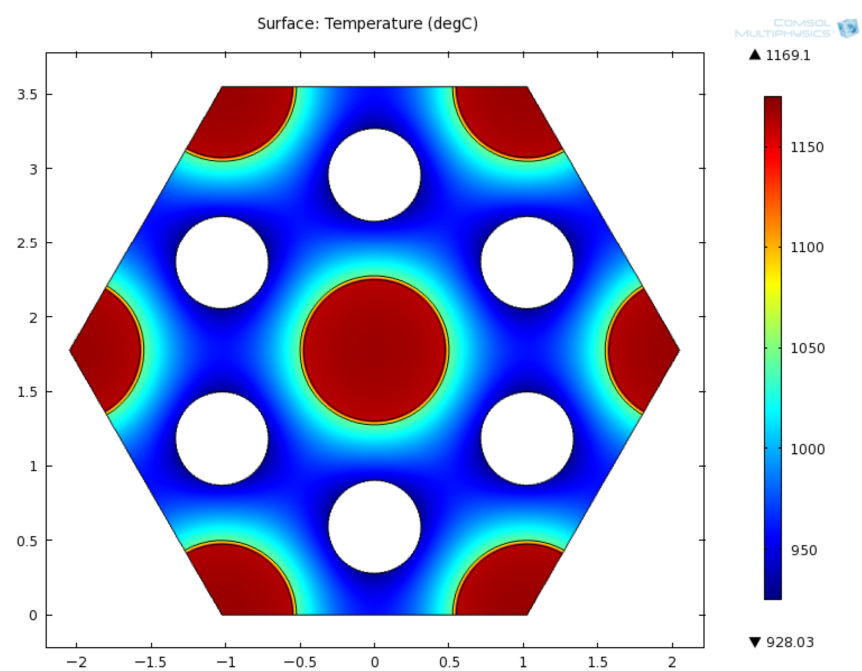
High Temperature Test Facility
Oregon State University

- **Prismatic Core "Fuel Element"**
  - Graphite electrodes embedded in ceramic block
    - Designer ceramic used to scale thermal conductivity of graphite moderator





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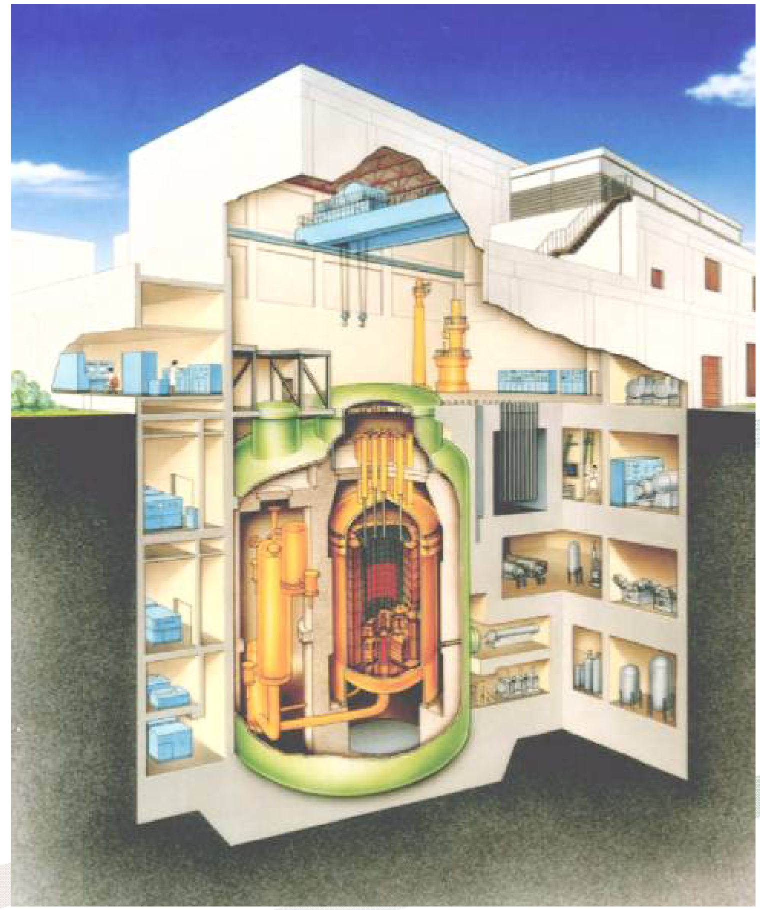
OECD HTTR-LOFC Program

- **High Temperature Engineering Test Reactor (JAEA)**

Major Specifications

Thermal power	30 MW
Fuel	Coated fuel particle / Prismatic block type
Core material	Graphite
Coolant	Helium
Inlet temperature	395 °C
Outlet temperature	950 °C (Max.)
Pressure	4 MPa



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OECD HTTR-LOFC Program

- **Proposed OECD Program**
  - Series of 3 Loss-of-Forced-Circulation Tests
    - Initial Conditions

Case no.	Reactor power (MWt)	Reactor inlet/outlet temp. (℃)	Flow rate (ton/h)
Run 1	9	180 / 320	45(rated)
Run 2	30 (full power)	395 / 850	45(rated)
Run 3	9	180 / 320	45(rated)

    - Test Procedure
      - All Tests:
        - All gas circulators tripped ⟹ no flow in primary
        - Reactor is not scrammed ⟹ simulated ATWS
      - Run #3:
        - All VCS pumps tripped ⟹ RCCS not active

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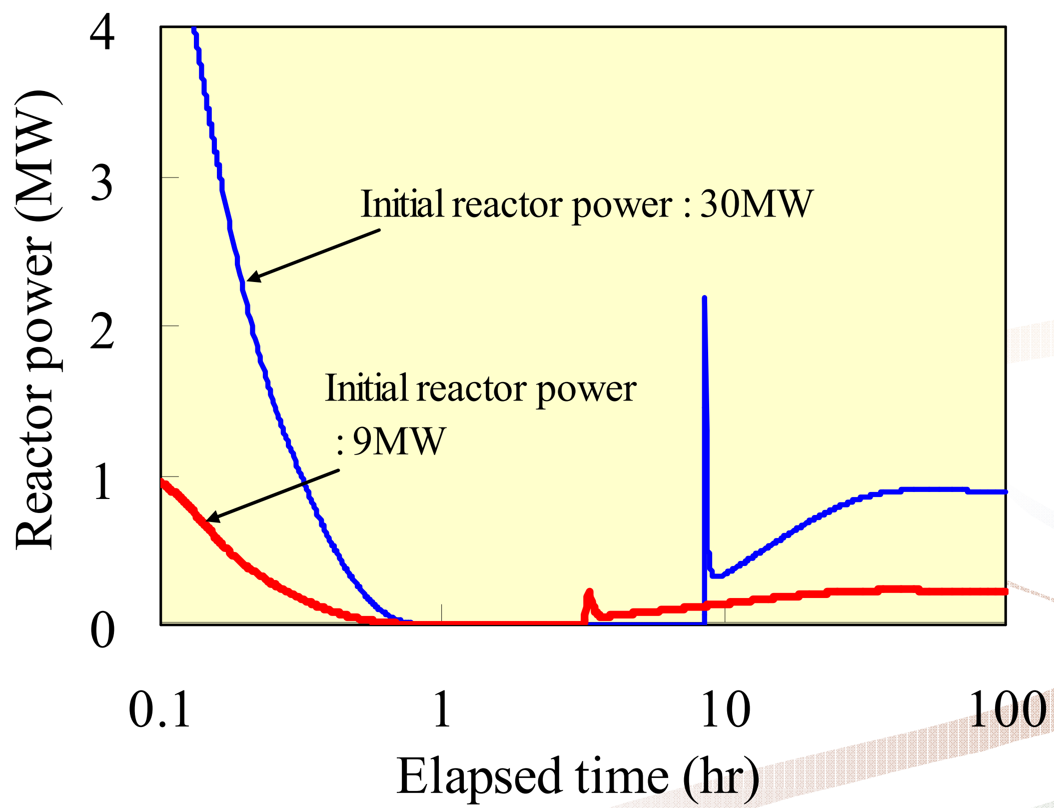
OECD HTTR-LOFC Program

Long term transient of reactor power during the loss-of-forced cooling tests



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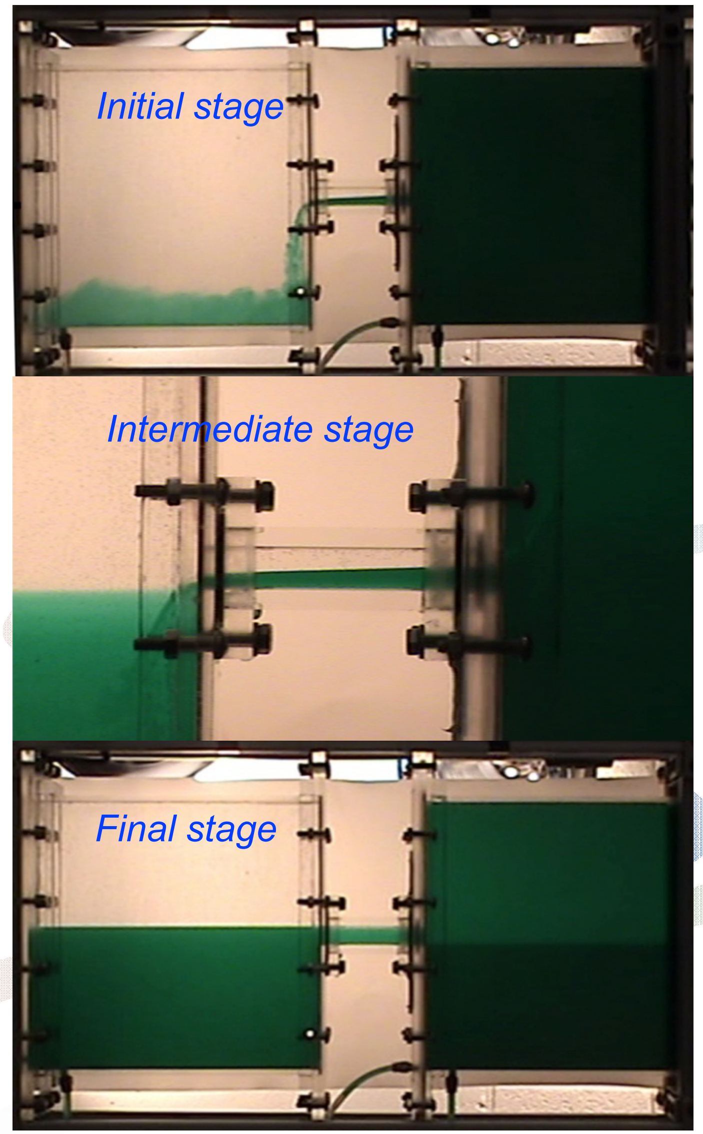
PBR Flow & Heat Transfer:
Texas A&M University

- **Areas of Investigation**
  - Pebble-Bed pressure drop
    - Annular & Cylindrical beds
  - Radial porosity distribution
    - Matched index of refraction (MIR) paired with laser induced fluorescence (LIF)
  - Radial velocity (flow) profile
    - Wall bypass effect
  - Pebble convective heat transfer
    - Induction heating of single pebble
      - HTC as function of Reynolds no. and proximity to the wall

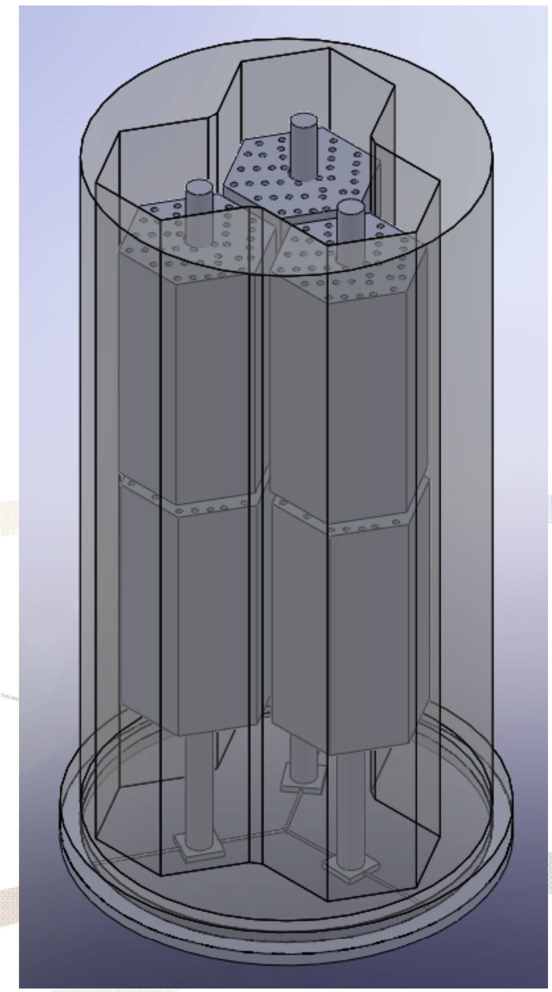
Air Ingress Flow Tests
Penn State University

- **Counter-Current Stratified Flow**
  - Study geometric effects on air-ingress rate:
    - Break orientation
    - L/D effect
    - Break geometry
  - Scoping studies with water-brine (complete)
  - Scaled experiments with He-Air (underway)



Prismatic Core Bypass Flow Study
Texas A&M University

- **Bypass Flow Measurements**
  - Matched index of refraction (MIR) coupled with particle image velocimetry (PIV) for flow measurement
  - Pressure drop measurements
    - Bypass gap
    - Coolant channel
  - Test section
    - 3 columns stacked 2 blocks high
    - Variable bypass gap widths
    - Variable cross-flow gap widths
      - Plane and wedge-shaped gaps



Acronym List

ATWS	Anticipated Transient Without Scram
CRDM	Control Rod Drive Mechanism
DCC	Depressurized Conduction Cooldown
FTIR	Fourier Transform Infrared
HTTF	High Temperature Test Facility
HTTR	High Temperature Engineering Test Reactor
JAEA	Japan Atomic Energy Agency
LIF	Laser Induced Fluorescence
LOFC	Loss of Forced Circulation
MHTGR	Modular High Temperature Gas Reactor (General Atomics design)
MIR	Matched Index of Refraction
NGNP	Next Generation Nuclear Plant

Acronym List

OD	Outside Diameter
OECD	Organization for Economic Cooperation and Development
OSU	Oregon State University
PBR	Pebble Bed Reactor
PCC	Pressurized Conduction Cooldown
PIV	Particle Image Velocimetry
PMR	Prismatic Modular Reactor
PSU	Penn State University
RCCS	Reactor Cavity Cooling System
TAMU	Texas A&M University
UW	University of Wisconsin
VCS	Vessel Cooling System

Back up slides

High Temperature Test Facility
Oregon State University

- **Background**
  - For first of a kind plants, NRC has performed confirmatory research in an integral test facility
    - Focus on passive system in B-DBA events looking for “cliffs” in behavior to ensure margin exists. Examples:
      - AP-600/1000: ROSA-IV and OSU/APEX
      - ESBWR: PUMA facility at Purdue
  - Cooperative Agreement (OSU/TAMU/UM)
    - Provides an opportunity to conduct this research for the NGNP in both a cost effective and timely manner.
    - Also provides aid in the development and validation of the NRC’s NGNP evaluation model.

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High Temperature Test Facility
Oregon State University

- **Status**
  - Procurement: **Contract**
    1. Vessel & balance-of-plant March 2011
    2. Data acquisition & control system May 2011
    3. Vessel internals, core & heaters June 2011
    4. Installation & conformance testing June 2011
  - Fabrication & installation: April-Aug. 2011
  - Conformance Testing: Sept.-Oct. 2011
  - Shakedown Testing: Nov. 2011-May 2012
  - Matrix Testing: June 2012-Sept. 2013

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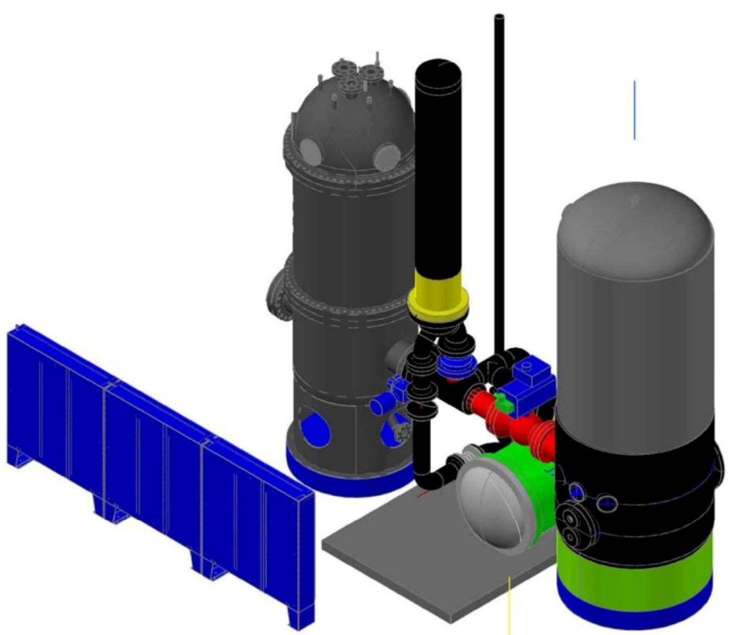
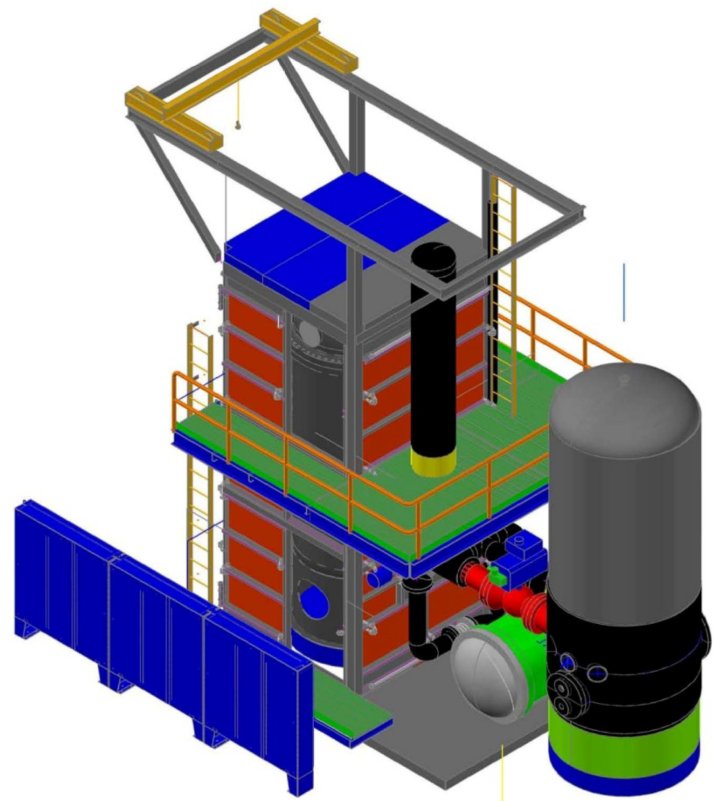
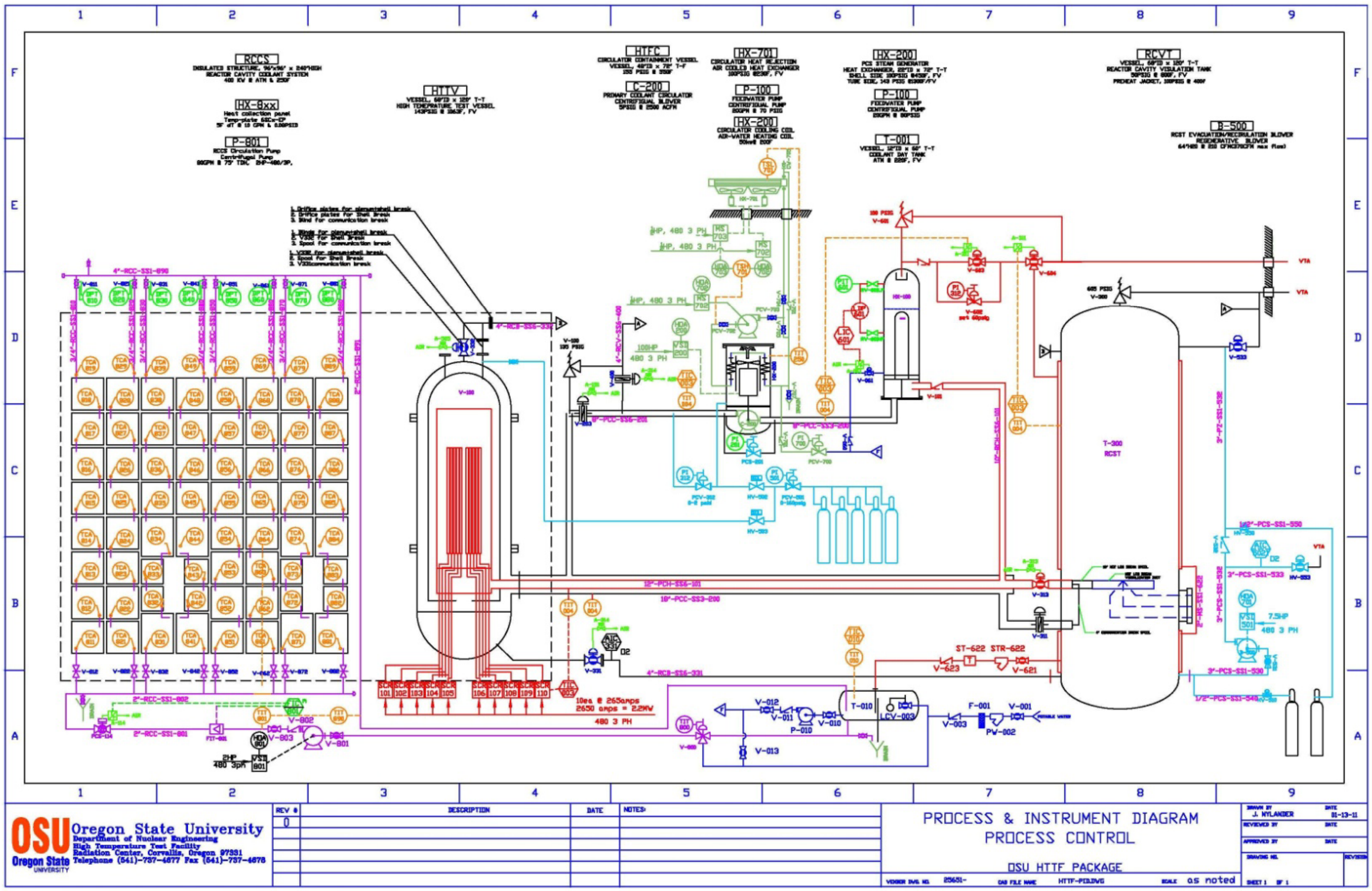
High Temperature Test Facility
Oregon State University

- **Facility overview**





High Temperature Test Facility
Oregon State University



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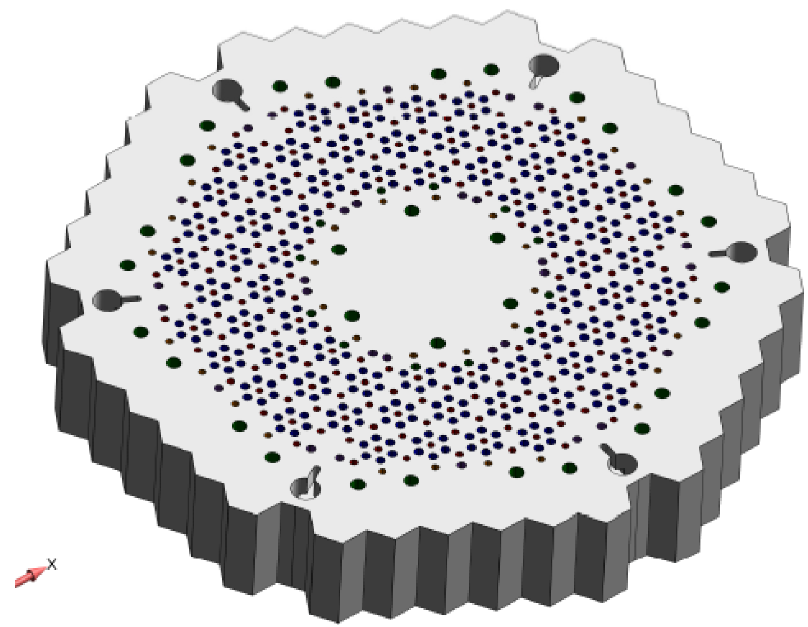
High Temperature Test Facility
Oregon State University

- **Core plate design**
  - Axial stack of 10 plates comprise heated core region
    - 210 heater rods
    - 516 coolant channels
    - 6 inner reflector control rod drive flow holes
    - 42 side reflector flow holes to simulate bypass flow



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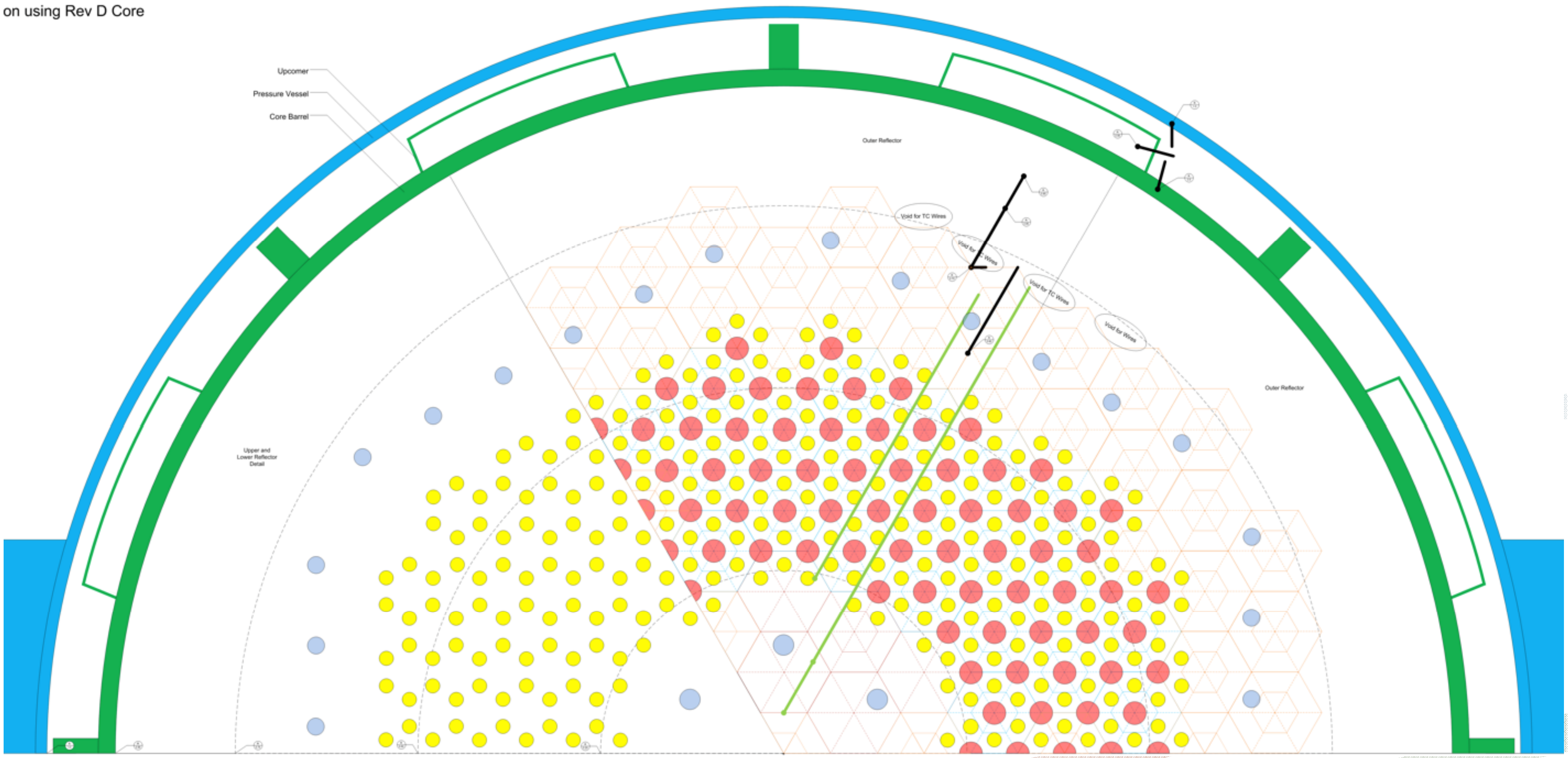
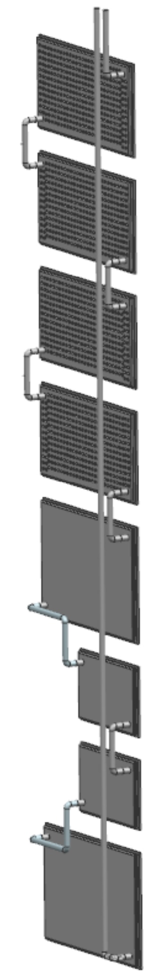
High Temperature Test Facility
Oregon State University

on using Rev D Core



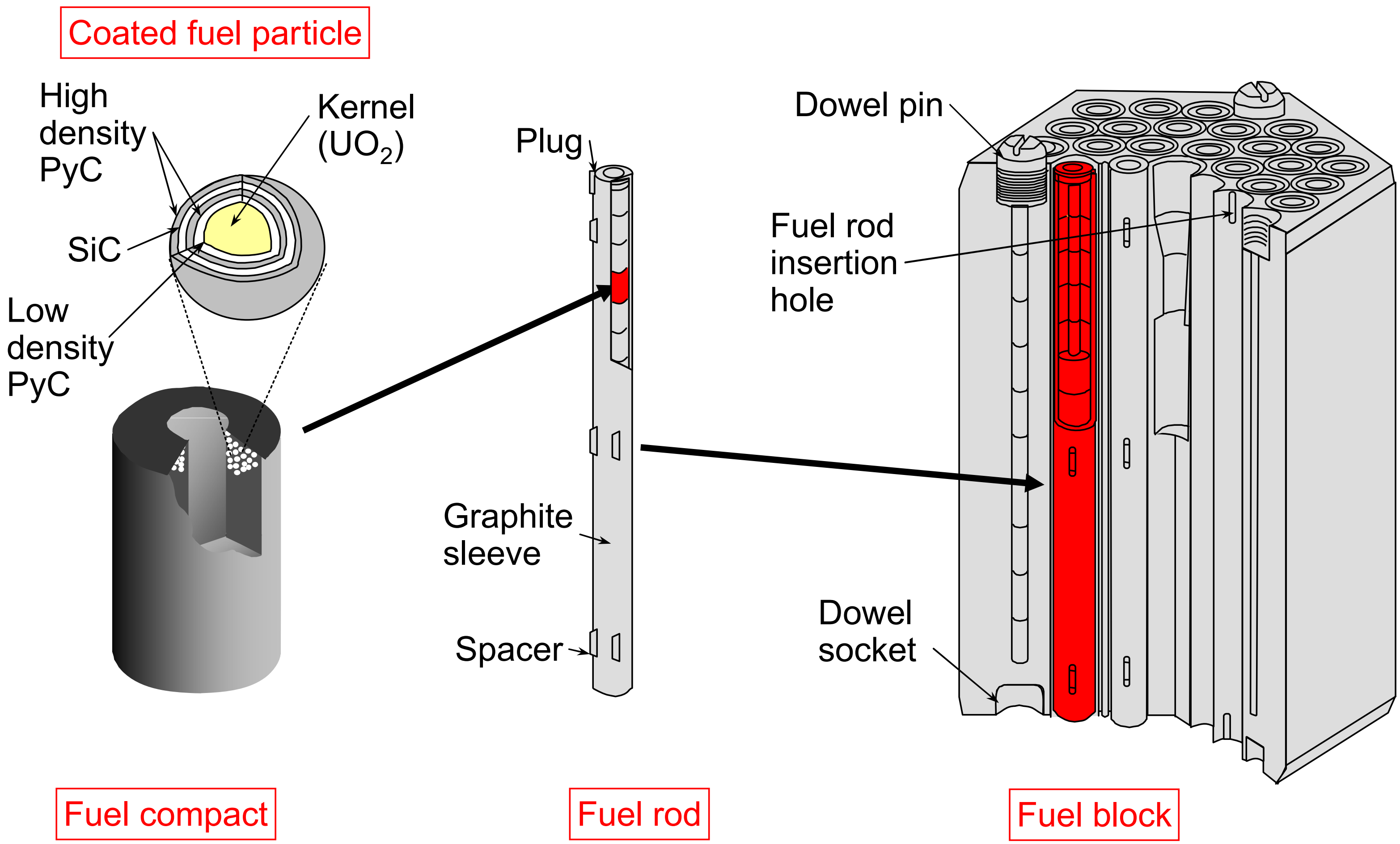
High Temperature Test Facility
Oregon State University

- **RCCS**
  - Water-cooled:
    - subcooled forced flow
  - 2x4 panels surround vessel
    - Each panel has 8 plate and tube heat exchangers
    - Inlet and outlet temperatures measured for every plate
      - Calorimetry provide axial distribution of heat flux



OECD HTTR-LOFC Program

- **HTTR Fuel Block & Fuel Rod**



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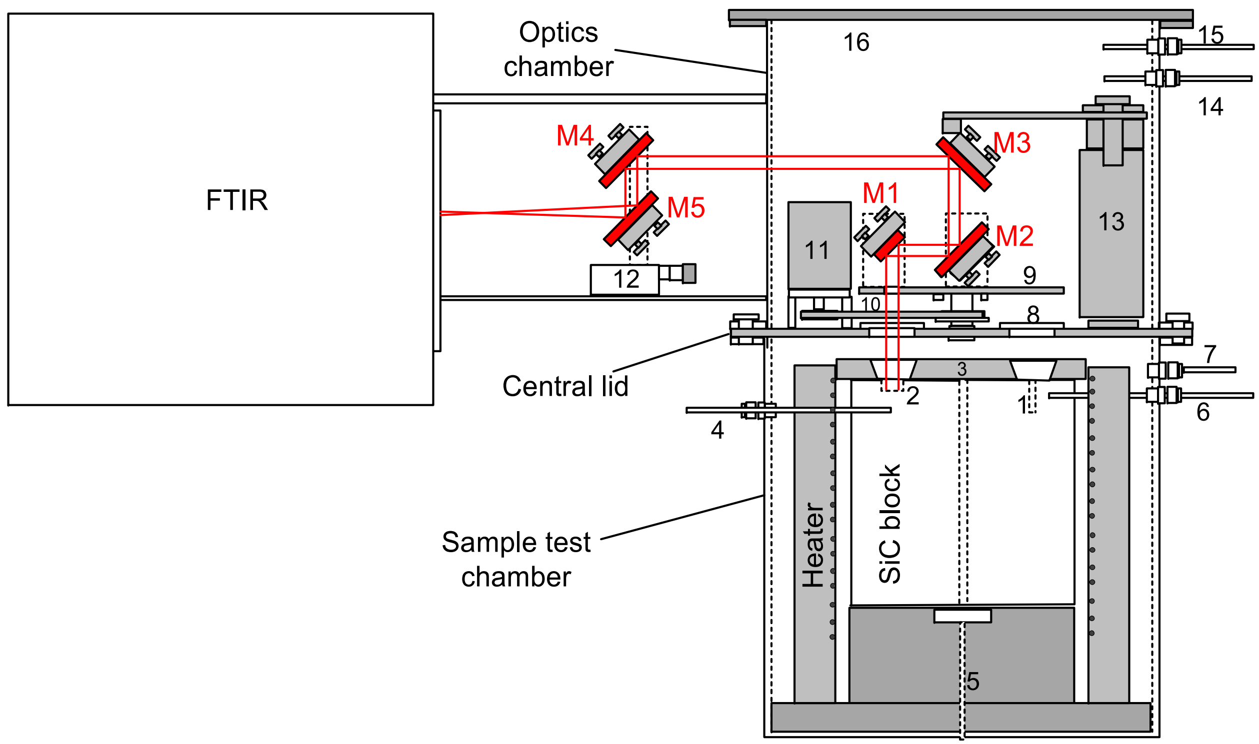
Component Emissivity
University of Wisconsin

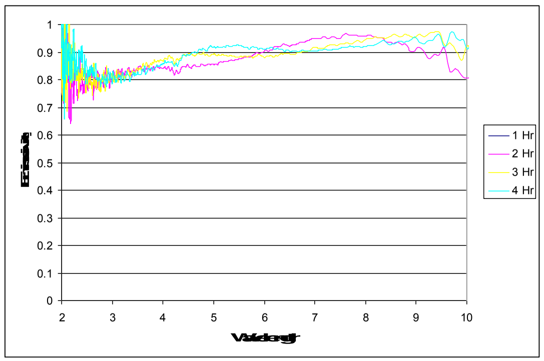
- **High Temperature Spectral Emissivity Measurement System**

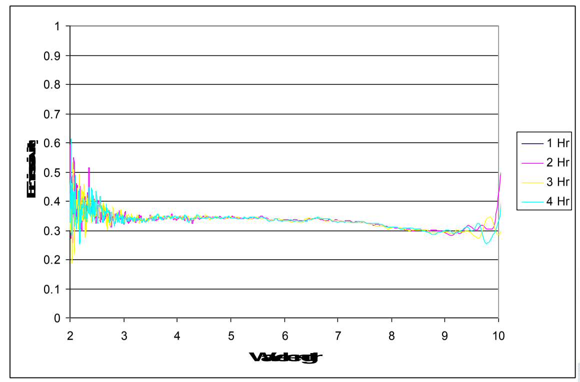


Component Emissivity
University of Wisconsin

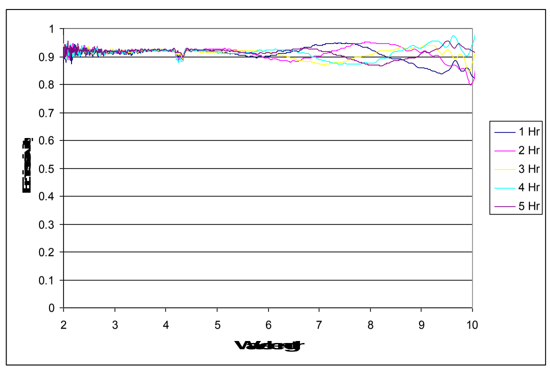
- **High Temperature Emissivity Measurement System**
  - Measurements for SA 508 and 316 Stainless

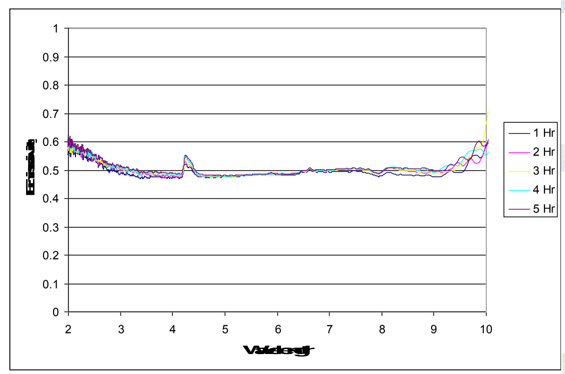
500ºC (in air)





700ºC (in air)





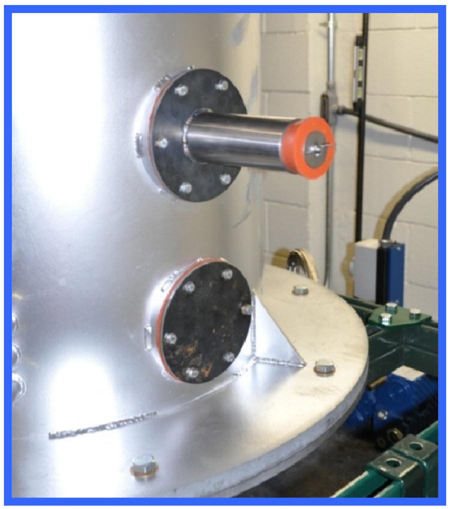
SA 508

316 Stainless Steel

Air Ingress Flow Tests
Penn State University

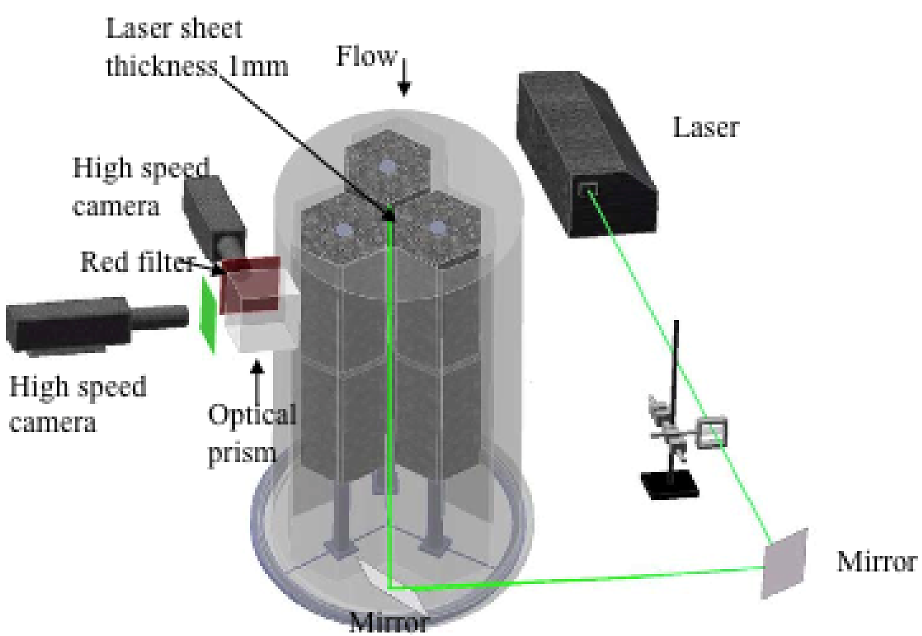
- **Helium-Air Test Facility**

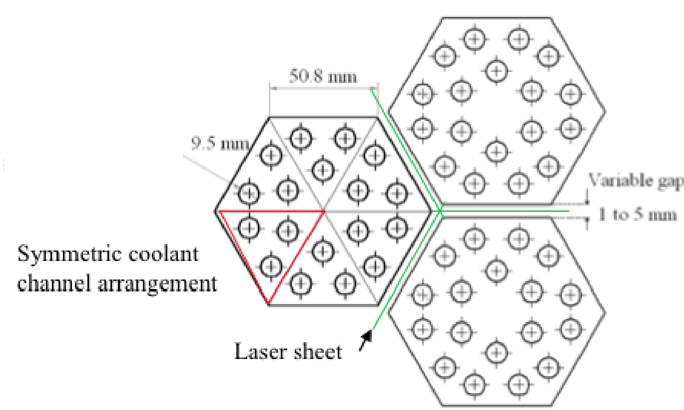
Horizontal Break



PMR Bypass Flow Study
Texas A&M University

- **Bypass flow data for code validation**
  - Matched index of refraction (MIR) paired with particle image velocimetry (PIV)





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Advanced Reactor Research Plan and Status for Graphite Materials

Dr. Makuteswara Srinivasan
Senior Materials Engineer
Corrosion and Metallurgy Branch
Division of Engineering
Office of Nuclear Regulatory Research

ACRS Future Plant Designs Subcommittee

May 13, 2011

Presentation Plan

- Objectives
- Graphite – Key Technical Issues
- Strategy for Graphite Research
- Graphite Core Component Safety Evaluation
- Codes and Standards – Current Status
- Safety Significant Graphite Phenomena
- Current NRC Research
- Next Steps

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Graphite Research Objectives

- **Establish independent technical bases for regulatory and safety decisions on graphite materials used in gas cooled reactors; address uncertainty in graphite behavior**
- **Use research results to confirm materials specifications, codes and standards, and to provide information and data for NRC's EM and PRAs**

The graphite research objectives and plan have been coordinated with input from NRO/DE, NRO/ARP, RES/DSA and RES/DRA staff.

Graphite Key Technical Issues Research Plan

- Availability and applicability of national codes and standards for graphite components
- Effects of impurities, including oxygen, on component degradation
- Inspection of graphite core components
- Performance of graphite under high irradiation
- Methodology for prediction of irradiated graphite properties
- GCC Structural Integrity Assessment
- Oxidation of reflector-grade graphite, fuel pebble matrix graphite, and graphite dust

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Strategy for Graphite Research

- **NRC staff expects applicant to provide technical bases for evaluation of graphite core components design**
- **Staff participates in Idaho National Lab (INL)/Oak Ridge National Lab (ORNL) and other DOE NGNP research to provide technical input on data needs**
- **Staff participates in codes and standards and domestic/international topical meetings**
- **Staff participates in domestic/international experimental programs and performs independent evaluation/interpretation of data**
- **If needed, staff conducts confirmatory research in specific areas**

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Challenges For Safety Evaluation of NGNP Graphite Core Components

- Develop NRC staff expertise, technical tools, and data for:
  - Materials performance analysis codes
  - Structural and component integrity analysis codes
  - Surveillance requirements and inspection codes
  - Tools to evaluate the efficacy of component degradation management programs
- Develop guidance documents for:
  - Safety review of graphite core components
  - In-service inspection and surveillance plans and techniques

Development of Codes and Standards for Graphite Components

- Since 2002, staff has actively participated on ASME and ASTM nuclear graphite committees and provided data need input for regulatory review of:
  - Design code for graphite core components
  - Standard specification for nuclear grade graphite core components
- Staff provided perspectives on graphite properties and recommended new standards development

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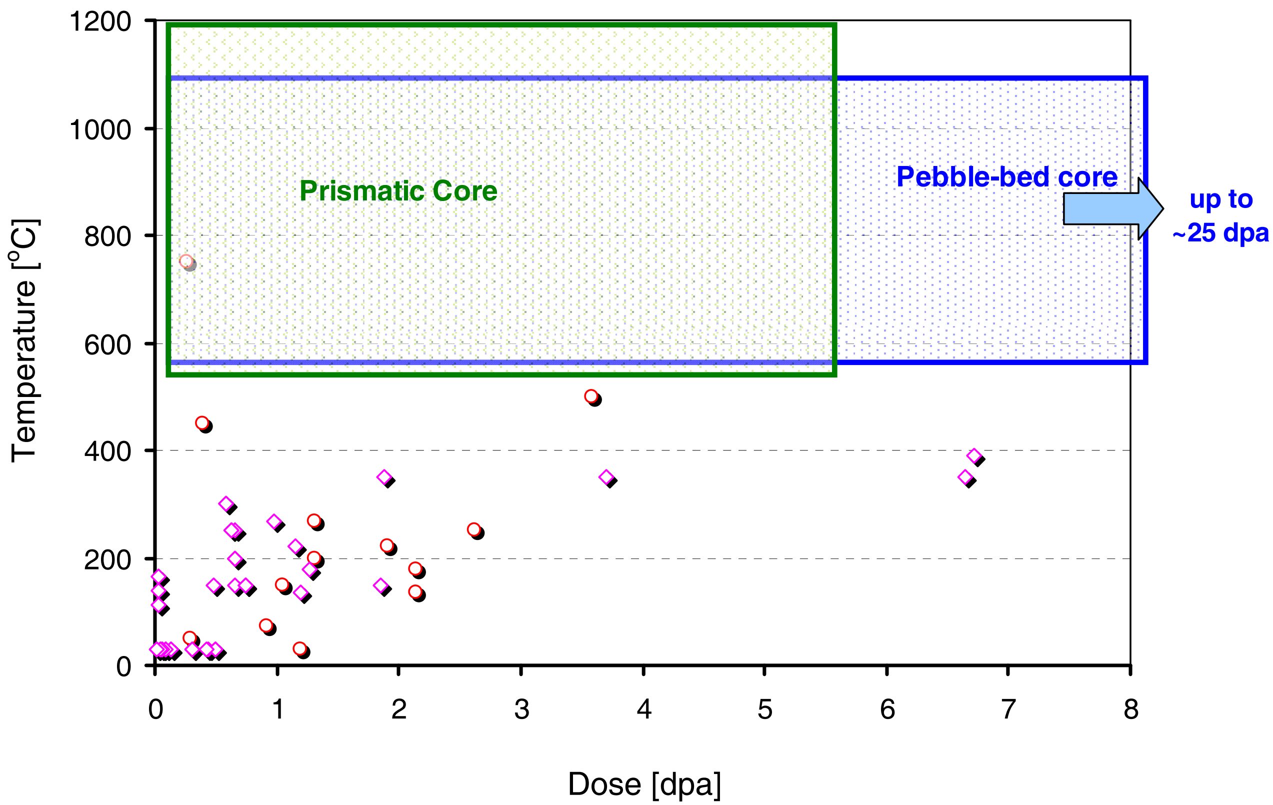
Safety Significant Graphite Phenomena

- During 2007, the NRC in cooperation with DOE, conducted a PIRT exercise to identify safety-significant graphite phenomena
- Many of the high-importance/low-knowledge phenomena are being addressed either by DOE research or by international research
- External research is expected to provide adequate information for regulatory needs for phenomena ranked as high importance/medium knowledge
- Staff will continue to provide technical input to DOE research regarding information needs, potential uncertainty in data and material/inspection/structural integrity assessment models

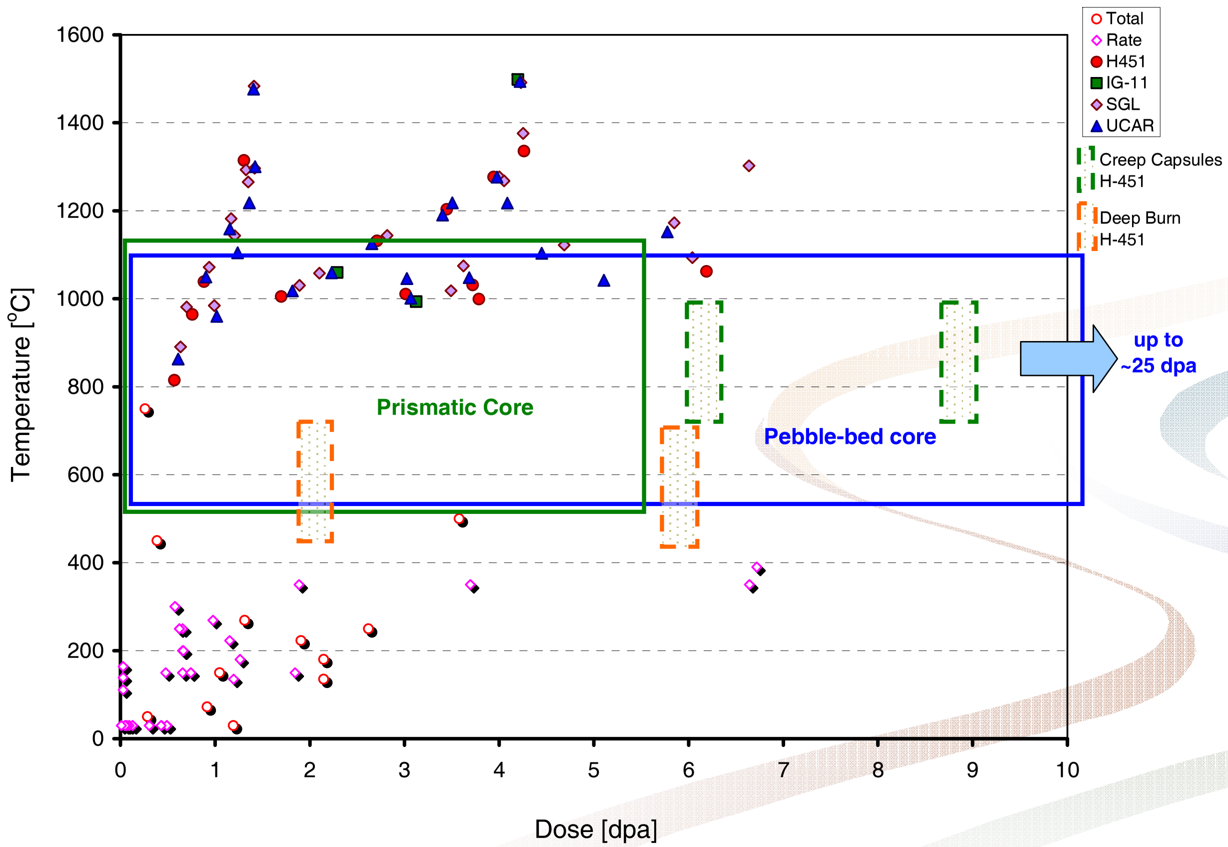
Current NRC Graphite Research

- "Nuclear Graphite Stored Energy Release," (2010- 2013) ORNL
  - Determine the safety significance of the energy stored due to high-temperature irradiation and released on subsequent heating to higher temperature.



Current NRC Graphite Research

- **"Nuclear Graphite Stored Energy Release," (2010- 2013) ORNL (Contd..)**
  - **Determine the safety significance of the energy stored due to high temperature irradiation and released on subsequent higher temperature heating.**

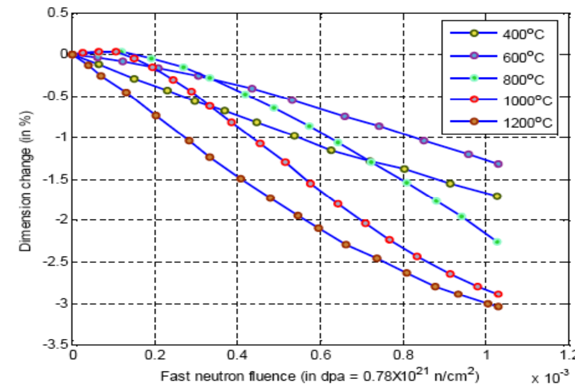


Current NRC Graphite Research

- "Nuclear Graphite Core Component Stress Analysis," (2010-2013) Argonne National Laboratory (ANL)
  - Develop a finite element stress analysis tool to independently verify and confirm applicant's design stresses and margins in graphite core component

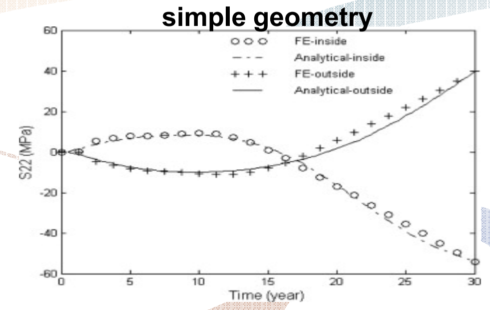
- The finite element code ABAQUS was chosen for conducting the heat conduction and stress analyses of graphite core components
- A procedure has been developed to incorporate the graphite properties into the ABAQUS code
- A user defined subroutine UMAT is being implemented for taking into account the interactive thermal and dimensional change strains due to irradiation creep

Phenomenological (empirical) material model has been developed for irradiated properties of



H-451 dimensional change, Ref: Graphite Design Handbook, DOE-HTGR-88111, General Atomics, September, 1988.

Reference graphite brick of simple geometry



Li H, Fok S L, and Marsden B J. "An analytical study on the irradiation-induced stresses in nuclear graphite moderator bricks". Journal of Nuclear Materials. Vol. 372. Issue 2-3. pp 164-170 (2008).

Other Graphite Issues

- **The staff will monitor research on:**
  - **In-service inspection of core graphite components**
  - **Oxidation under reactor normal operating conditions (diffusion-controlled oxidation)**
  - **Loss of graphite under simulated air- and water-ingress events**
  - **International reactor accident experience (Windscale and Chernobyl)**
  - **Tribology aspects in graphite, fuel matrix abrasion, and dust generation**
  - **Potential for dust oxidation/explosion**
- **The staff expects the applicant to provide complete information on these and other subjects which may emerge**

Next Steps

- Monitor DOE research and provide technical input as appropriate
- Complete research at ORNL and ANL and document findings
- Continue participation in codes and standards organizations, and monitor graphite research
- Monitor international graphite research related to graphite irradiation and irradiation creep
- During 2013, begin developing regulatory guides and specific standard review plan additions for review of graphite core component design

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Abbreviations

Abbreviation	Stands For
ABAQUS	A Finite Element (Stress) Analysis Software
AGR	Advanced Gas Reactor
ANL	Argonne National Laboratory
ARP	Advanced Reactor Program
ASME	American Society for Mechanical Engineers
ASTM	American Society for Testing Materials
BNL	Brookhaven National Laboratory
DOE	U.S. Department of Energy
dpa	Displacement per atom, a unit of cumulative damage measure
DRA	Division of Risk Analysis
DSA	Division of Systems Analysis
EM	Accident Analysis Evaluation Model
GCC	Graphite Core Component

Abbreviations (Contd..)

Abbreviation	Stands For
HTGR	High Temperature Gas Cooled Reactor
HTTR	High Temperature Test Reactor
INL	Idaho National Laboratory
ISI	In-Service Inspection
ISO	International Standards Organization
LWR	Light Water Reactor
NGNP	Next Generation Nuclear Plant
NRO	Office of New Reactors
ORNL	Oak Ridge National Laboratory
PCRV	Pre-stressed Concrete Reactor Vessel
PIRT	Phenomenon Identification and Ranking Table
PRA	Probabilistic Risk Assessment/Analysis
UMAT	A Subroutine Software of Materials Properties

Backup Slides

In-service Inspection of Graphite Core Components

- **The staff will monitor development in ISI for gas cooled reactors**
- **ISI depends on component design for specific reactor type**
- **Inspection intervals are less frequent compared to LWRs**
- **ASME Sec XI code case for HTGR yet to be developed**
- **Most operating experience from British AGRs and Magnox reactors**
  - **Channel bore measuring unit (CBMU) for assessing channel distortion**
  - **Remote camera visual inspection**
  - **Trepanning to test and assess brick properties degradation**
  - **Emerging development of ex-reactor eddy current technique**
- **Limited experience from Japanese HTTR**
  - **Application of remote video surveillance**
  - **Application of acoustic emission technique**
  - **Research on ex-situ measurement of internal stresses**

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"Burning" of Graphite

- **Burning implies a self sustaining reaction. In all practical circumstances, graphite burning by this definition is not possible**
- **However, given a constant supply of oxygen, graphite will react resulting in graphite loss – this is a well-studied phenomenon**
  - **A major industrial use for electrode graphite – for electric arc furnace for steel melting – involves consumable electrodes**
- **Graphite – oxygen reaction increases with increased temperature starting to become significant around 475 °C; however, for solid graphite blocks, the reaction is limited by the by-products of the oxidation process**
  - **After the Chernobyl accident, an experiment was performed at the Kurchatov Institute. A graphite block was heated to 1000 °C and left to stand in air. The graphite block just cooled down without significant oxidation**
  - **During graphite manufacture, it is not unusual to see graphite blocks glowing red waiting to cool in air**

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Glow of Graphite Electrodes in Steel Production

Figure 1. Graphite Production Longitudinal Furnace

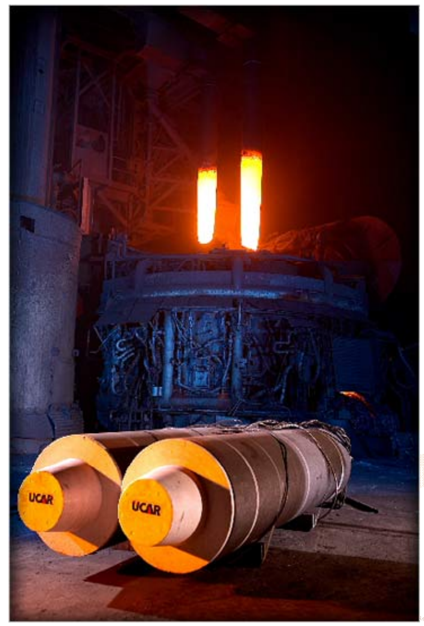

Figure 2. Graphite Electrode Glow In Arc Furnace

Figure 3. Three graphite electrodes glow red hot after their removal from an electric arc furnace used to produce steel.

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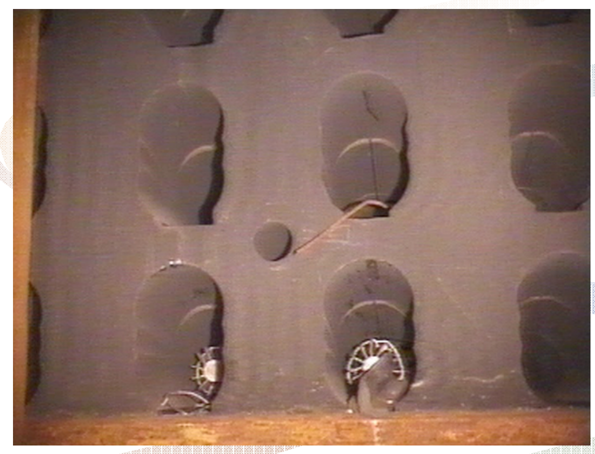
"Burning" of Graphite - Continued

- The amount of material lost is a function of density and graphite purity and grade – irradiated graphite can lose material faster and in greater amounts than non-irradiated graphite, though this effect is small
- Despite long duration (over 24 hours) and the intense heat (> 1200 °C) in the Windscale fire (which was a uranium metal fire), the amount of graphite oxidized was relatively small
  - Inspections of Windscale pile have shown that there was NOT a graphite fire: damage to graphite, caused by severely overheated fuel assemblies, was localized. (Ref: Nuclear Safety Advisory Committee Meeting of RG2 with Windscale Pile 1 Decommissioning Project Team-29/09/2005)
  - The heat from the uranium metal fire enlarged the diameter of the fuel channels, joining the adjacent channel in a few places. At the back of the core where the oxygen was depleted, there was little visible damage
- The NRC has studied this phenomenon after the Chernobyl rector accident

Before



After

Photos courtesy of Professor Barry Marsden of University of Manchester, March 2011.

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“Burning” of Graphite

Accident	Fuel/Cladding	Coolant	Burning/Fire Due to	Oxidation Core Graphite Loss
Windscale (1957)	Uranium/Aluminum cladding	Air	Uranium metal	Not significant
Chernobyl (1986)	UO_2/Zircaloy cladding	Water	Zircaloy	No data; estimated to be significant

NGNP HTGR is expected to use ceramic coated fuel particles with higher temperature capability, and helium gas as the coolant.

NGNP Fuel project is testing two TRISO fuel types in the ATR
– 500 µm UO_2 kernel: used in AREVA pebble bed design
– 435 µm UCO kernel: used in GA prismatic design

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Mid 1970s: Combustion Hazard in HTGR

- H.B. Palmer, M. Sibulkin, R.A. Strehlow, and C.H. Yang, "An Appraisal of Possible Combustion Hazards Associated With A High-Temperature Gas-Cooled Reactor," BNL-NUREG-50764, March 1978
    - Studied combustion hazards resulting from a primary coolant circuit depressurization followed by water or air ingress into the prestressed concrete reactor vessel (PCRV)
    - Studied reactions between graphite and steam or air which produce the combustible gases H_2 and/or CO; when mixed with air in the PCRV, flammable mixtures may be formed
    - Possible circumstances leading to these hazards and the physical characteristics related to them were delineated and studied

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Mid 1980s: Assessment of Potential for Graphite Fire in U.S. Reactors

- D.G. Schweitzer, D.H. Gurinsky, E. Kaplan, and C, Sastre, "A Safety Assessment of the Use of Graphite in Nuclear Reactors Licensed by the U.S. NRC," NUREG/CR-4981 (BNL), September 1987
  - Evaluated the potential for graphite fire in U.S. research reactors and Fort St. Vrain reactor
  - Considered stored energy contributions
  - Established necessary conditions of geometry, temperature, oxygen supply, reaction product removal, and a favorable heat balance for self-sustained rapid graphite oxidation
  - Concluded that no credible potential for a graphite burning accident in reactors analyzed

Early 2010: Graphite Dust Explosion Research

- Available information to date indicates that disruptive graphite dust explosion is highly unlikely. The available heat content and the minimum required ignition energy for non-irradiated graphite dust and dust from irradiated graphite do not favor explosion, compared to organic material vapor/gas and inorganic material dust cloud
- Studies were conducted in Italy, France, and Great Britain to inform decommissioning of graphite reactors
  - A. Wickham and L. Rahmani, conducted (EPRI-sponsored) a review of this work. (IAEA TECDOC-1647, Oct 2010)
    - Nuclear grade graphite dust is formally classified as “weakly explosible”, based on results from an ISO standard or equivalent standard test. These standards require ignition of the material after suspension into a confined and known volume, utilizing chemical igniters of energy typically 10–20 kJ.
- Thermal lances were used to cut the restraints of Windscale reactor during decommissioning. Even though there was dust around, and the restraints were directly in contact with the graphite, there was no fire or explosion. (B. Marsden, March 2011 communication to M. Srinivasan)
- Dust is invariably created during graphite manufacture, fabrication of parts, and machining
  - There has not been a recorded dust explosion event in more than 100 years of graphite manufacture

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Typical Graphite Fabrication Shop

Source: UCAR Carbon Company, “The Industrial Graphite Engineering Handbook”,

Relative Comparison of Ignition Energies for Explosibility

FUEL - Liquid (vapor or mist), gas, or solid capable of being oxidized. Combustion always occurs in the vapor phase; liquids are volatized and solids are decomposed into vapor prior to combustion.

OXIDANT - A substance which supports combustion –Usually oxygen in air.

IGNITION SOURCE - An energy source capable of initiating a combustion reaction.

ATMOSPHERE	MATERIAL	MINIMUM IGNITION ENERGY, JOULES
VAPOR/GAS	PROPANOL	0.00065
	ETHYLE ACETATE	0.00046
	METHANE	0.00028
	PROPANE	0.00025
	ETHANE	0.00024
	METHANOL	0.00014
	ACETYLENE	0.000017
	HYDROGEN	0.000016
	CARBON DISULPHIDE	0.000009
DUST CLOUD	PVC	2
	ZINC	0.2
	WHEAT FLOUR	0.05
	POLYETHYLENE	0.03
	SUGAR	0.03
	MAGNESIUM	0.02
	SULPHUR	0.015
	ALUMINIUM	0.01
	EPOXY RESIN	0.009
	ZIRCONIUM	0.005
	NUCLEAR GRAPHITE	10,000-20,000

Note: The ignition energy depends on particle size distribution and other factors; but, a general comparison is still instructive.

Ref: E. Ebadat, Dust Explosion Hazard Assessment Including OSHA Combustible Dust NEP", Presentation at2009 Annual OSHA & Workshop Safety Conference, May 19, 2009 for all materials, except nuclear graphite. A. Wickham and L. Rahmani, IAEA TECDOC-1647, Oct 2010, for nuclear graphite.

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Chemical Composition of Coals

Name	Volatiles %	Carbon %	Hydrogen %	Oxygen %	Sulfur %	Heat content kJ/kg
Braunkohle (Lignite)	45-65	60-75	6.0-5.8	34-17	0.5-3	<28470
Flammkohle (Flame coal)	40-45	75-82	6.0-5.8	>9.8	~1	<32870
Gasflammkohle (Gas flame coal)	35-40	82-85	5.8-5.6	9.8-7.3	~1	<33910
Gaskohle (Gas coal)	28-35	85-87.5	5.6-5.0	7.3-4.5	~1	<34960
Fettkohle (Fat coal)	19-28	87.5-89.5	5.0-4.5	4.5-3.2	~1	<35380
Esskohle (Forge coal)	14-19	89.5-90.5	4.5-4.0	3.2-2.8	~1	<35380
Magerkohle (Non baking coal)	10-14	90.5-91.5	4.0-3.75	2.8-3.5	~1	35380
Anthrazit (Anthracite)	7-12	>91.5	<3.75	<2.5	~1	<35300
Percent by weight						

The classification of coal is generally based on the content of volatiles. However, the exact classification varies between countries. The coal data, presented here, is German coal classification and is from "Eberhard Lindner; Chemie für Ingenieure; Lindner Verlag Karlsruhe, S. 258", as cited in several sources of coal information in the web. Note that, graphite, after greater than 2800 °C graphitization, has just carbon as its chemical constituent. That is, there are no volatiles, hydrogen, and oxygen (fuel source) so sustained burning in the presence of limited oxygen and ignition source is not favored. Comparatively, the heat content of 30 °C irradiated nuclear graphite is less than 3,000 kJ/kg, and that for 450 °C irradiated graphite is less than 75 kJ/kg. This heat content can be considered to be negligible for combustion.

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