UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

March 8, 2013

MEMORANDUM TO: ACRS Members

FROM: Mark L. Banks, Senior Staff Engineer ***/RA/***
Technical Support Branch

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING – REVIEW OF NEXT GENERATION NUCLEAR PLANT RESEARCH AND LICENSING ISSUES, JANUARY 17, 2013, ROCKVILLE, MARYLAND

The minutes for the subject meeting were certified on March 8, 2013, as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: Certification Letter
Minutes
Meeting Transcript

cc w/o Attachment: E. Hackett
C. Santos

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MEMORANDUM TO: Mark L. Banks, Senior Staff Engineer
Technical Support Branch, ACRS

FROM: Dr. Dennis C. Bley, Chairman
Future Plant Designs Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING – REVIEW OF NEXT GENERATION NUCLEAR PLANT RESEARCH AND LICENSING ISSUES, JANUARY 17, 2013, ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on January 17, 2013, are an accurate record of the proceedings for that meeting.

__________***/RA/***_________________

Dr. Dennis C. Bley, Chairman Date: March 8, 2013
Future Plant Designs Subcommittee

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING MINUTES
JANUARY 17, 2013
ROCKVILLE, MARYLAND

INTRODUCTION

The Advisory Committee on Reactor Safeguards (ACRS) Future Plant Designs Subcommittee met in room T-2B1at the Headquarters of the U.S. Nuclear Regulatory Commission (NRC), located at 11545 Rockville Pike, Rockville, Maryland, on January 17, 2013. The Subcommittee was briefed by representatives of the U.S. Department of Energy (DOE) and Idaho National Laboratory (INL) regarding research and licensing issues pertaining to DOE's Next Generation Nuclear Plant (NGNP) project. The INL presentations included key information contained in white papers submitted to the NRC staff.

The meeting convened at 8:30 AM and adjourned at 2:17 PM. The meeting was open to the public. No written comments were received from members of the public related to this meeting. Mr. Farshid Shahrokhi, AREVA US, representing the NGNP Industry Alliance, provided verbal comments during the meeting.

ATTENDEES

ACRS Members
Dennis Bley (Chairman)
Sam Armijo
Charles Brown
Michael Corradini
Harold Ray
Joy Rempe
William Shack
Thomas Kress (Consultant)
ACRS Staff
Maitri Banerjee (DFO)
Presenters
Don Carlson, NRC/NRO
Carl Sink, DOE
Fred Silady, INL
Mark Holbrook, INL
David Alberstein, INL
David Petti, INL
NRC Staff
Jim Shea, NRO
Shie-Jeng Peng, NRO
Arlon Costa, NRO
Thomas Boyle, NRO
Anna Bradford, NRO
Russell Chazell, NRO
Patricia Milligan, NSIR
Michelle Hart, NRO
Brian Thomas, NRO
Neil Ray, NRO
Eric Reichelt, NRO
Jonathon DeGange, NRO
Tarico Sweat, NRO
Richard Lee, RES
Nan Chien, NRO
Other Attendees
David Hanson, INL
Jim Kinsey, INL
Farshid Shahrokhi, AVEVA
Edward Burns, Westinghouse
George Zinke, Entergy
Janelle Zamore, DOE
John Kelly, DOE
Jessica Press-Williams, DOE
MA Feltus, DOE

SUMMARY

The purpose of this meeting was for the Future Plant Designs Subcommittee to receive an information briefing from the U.S. Department of Energy and its lead laboratory, Idaho National Laboratory (INL), on the Next Generation Nuclear Plant (NGNP) project. INL briefed the Subcommittee on the NGNP project's safety design approach and technology development focus. In addition, INL discussed the process used to select NGNP licensing basis events. The INL presentations included key information from the following submitted white papers currently under review by the NRC staff:

- NGNP Defense-in-Depth Approach
- NGNP Fuel Qualification
- HTGR Mechanistic Source Terms
- NGNP Licensing Basis Event Selection
- NGNP Structures, Systems, and Components Safety Classification
- Determining the Appropriate EPZ Size and Emergency Planning Attributes for an HTGR
- NGNP Probabilistic Risk Assessment
- Modular HTGR Safety Basis and Approach

At the conclusion of the meeting, the subcommittee members and their consultant commented on various aspects of the information presented by INL. Several expressed interest in having additional discussions regarding the concept of defense-in depth and how it relates to NGNP – one member was not sanguine with the concept of the "super" fuel particle that would never fail, while another was comfortable with the fuel particle being the major fission product barrier and that the traditional defense-in-depth concept of reactor coolant system and containment was not necessary. The need for the NRC staff to clearly document its positions on the key NGNP issues was emphasized so that any future HTGR work can benefit from the INL NGNP research and analysis. A comment commended the NGNP approach to licensing, as well as the NGNP fuel concept. Some concern to NGNP and the NRC's quantitative health objectives (QHOs) was expressed, specifically regarding the concept of multiplying the overall frequency by the number of modules at a site: will the number of modules be a factor in the determination of whether the site meets the prompt fatal QHO?

SIGNIFICANT ISSUES	
Issue	Reference Pages in Transcript
Historical HTGR major issues	7
Prismatic vs. pebble bed decision	12-13
Steam generators and reactor moisture monitoring	38-42
Emergency planning (2-hour timeframe)	48-50
Conceptual design vs. preliminary design	54-56
PRA – use of frequency	60-70
Emergency planning zones (EPZs)	85-87

Design basis accident formulation	95-100
Designed to meet the Protective Action Guidelines (PAGs) at EAB, response of multiple reactor modules to accidents	107-112
Fuel particle defects specification, heavy metal contamination, monitoring for fission product release and circulating activity	125-132
Off-normal events and radionuclide release mechanisms	134 -136
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Use of reactor building as functional containment and radionuclide release	146-152
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Variability of fuel particle SiC layer thickness and failure probability, superior radiation performance of NGNP fuel	211-215
Post irradiation examination (PIE)	215-231
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Action Item	Reference Pages in Transcript
Mr. Kinsey (INL) – provide additional information on defense-in-depth at the April 9 Subcommittee Meeting	142-144

DOCUMENTS PROVIDED TO THE SUBCOMMITTEE

Historic

1. U.S. NRC, NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor," March 1989 (ML052780497)
2. U.S. NRC Memorandum, "Draft Copy of Preapplication Safety Evaluation Report (PSER) for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)," February 26, 1996 (ML052780519)
3. U.S. NRC, SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationships to Current Regulatory Requirements," April 8, 1993 (ML040210725)
4. U.S. NRC, SRM-SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationships to Current Regulatory Requirements," July 30, 1993 (ML003760774)

5. U.S. NRC, SECY-98-300, “Options for Risk-Informed Revisions to 10 CFR Part 50 – Domestic Licensing of Production and Utilization Facilities,” December 23, 1998 (ML992870048)
6. U.S. NRC, SECY-03-047, “Policy Issues related to Licensing Non-Light-Water Reactor Designs,” March 28, 2003 (ML030160002)
7. U.S. NRC, SRM-SECY-03-047, “Policy Issues related to Licensing Non-Light-Water Reactor Designs,” June 26, 2003 (ML031770124)
8. U.S. NRC, SECY-04-157, “Status of Staff’s Proposed Regulatory Structure for New Plant Licensing and Potentially New Policy Issues,” August 30, 2004 (ML042370388)
9. U.S. NRC, SECY-05-006, “Second Status Paper on the Staff’s Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing,” January 7, 2005 (ML042370388)
10. U.S. NRC Policy Statement, “Safety Goals for Operations of Nuclear Power Plants,” August 4, 1986 (ML051580401)
11. U.S. NRC Policy Statement, “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities,” August 16, 1995 (ML021980535)
12. U.S. NRC Policy Statement, “Regulation of Advanced Nuclear Power Plants,” July 12, 1994 (ML051740661)

Recent NGNP Documents

1. U.S. NRC, SRM-SECY-08-0019, “Licensing and Regulatory Research Related to Advanced Nuclear Reactors,” June 11, 2008 (ML081630507)
2. U.S. NRC, COMSECY-08-0018, “Report to Congress on Next Generation Nuclear Plant (NGNP) Licensing Strategy,” May 12, 2008 (ML081330510)
3. U.S. NRC, SECY-11-052, “Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors,” October 28, 2011 (ML112570439)
4. Idaho National Laboratory, INL/EXT-11-22708, “Modular HTGR Safety Basis and Approach,” August 2011 (ML11251A169)
5. Idaho National Laboratory Letter, “Next Generation Nuclear Plant Submittal – Confirmation of Requested NRC Staff Positions,” July 6, 2012 (ML121910310)
6. Idaho National Laboratory, INL/EXT-10-17686, “NGNP Fuel Qualification White Paper,” July 2010 (ML102040261)
7. Idaho National Laboratory, INL/EXT-10-17997, “Mechanistic Source Terms White Paper,” July 2010 (ML102040260)
8. Idaho National Laboratory, INL/EXT-09-17139, “Next Generation Nuclear Plant Defense-in-Depth Approach,” December 2009 (ML093490191)
9. Idaho National Laboratory, INL/EXT-10-19521, “Next Generation Nuclear Plant Licensing Basis Event Selection White Paper,” September 2010 (ML102630246)
10. Idaho National Laboratory, INL/EXT-10-19509, “Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification White Paper,” September 2010 (ML102660144)
11. Idaho National Laboratory, INL/EXT-11-21270, “Next Generation Nuclear Plant Probabilistic Risk Assessment White Paper,” September 2011 (ML11265A082)

12. Idaho National Laboratory, INL/EXT-09-17187, “NGNP High Temperature Materials White Paper,” June 2010 (ML101800221)
13. Idaho National Laboratory, INL/EXT-10-19799, “Determining the Appropriate Emergency Planning Zone Size and Emergency Planning Attributes for an HTGR,” October 2010 (ML103050268)
14. U.S. NRC, “Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms,” February 12, 2012 (ML120240669)
15. U.S. NRC, “Assessment of White Paper Submittals on Defense-in-Depth, Licensing Basis Event Selection, and Safety Classification of Structures, Systems, and Components,” February 15, 2012 (ML120170084)

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards Future Plant Design Subcommittee
Docket Number:	(n/a)
Location:	Rockville, Maryland
Date:	Thursday, January 17, 2013
Work Order No.:	NRC-3037 Pages 1-245

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

FUTURE PLANT DESIGN SUBCOMMITTEE

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NGNP RESEARCH AND LICENSING ISSUES

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THURSDAY, JANUARY 17, 2013

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ROCKVILLE, MARYLAND

The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Dennis C. Bley, Chairman, presiding.

COMMITTEE MEMBERS:

DENNIS C. BLEY, Chairman

J. SAM ARMIJO, Member

CHARLES H. BROWN, JR. Member

MICHAEL L. CORRADINI, Member

HAROLD B. RAY, Member

JOY REMPE, Member

WILLIAM J. SHACK, Member

ACRS CONSULTANTS PRESENT:

THOMAS A. KRESS

NRC STAFF PRESENT:

MAITRI BANERJEE, Designated Federal Official

DON CARLSON, NRO

JIM SHEA, NRO

ALSO PRESENT:

DAVID ALBERSTEIN, INL

DAVID HANSON, INL

MARK HOLBROOK, INL

JIM KINSEY, INL

DAVID PETTI, INL

FARSHID SHAHROKHI, AREVA

FRED SILADY, INL

CARL SINK, DOE

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

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

(8:31 a.m.)

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

We have with us today ACRS members Doctors Armijo, Corradini, Rempe, Powers, Ray, and we expect Mr. Brown to join us later. Dr. Tom Kress is here as our consultant. And Ms. Maitri Banerjee of the ACRS staff is our designated Federal official for this meeting.

The purpose of today's meeting is to receive an information briefing from the Idaho National Laboratory Staff on the NGNP project. DOE, the official sponsor of the NGNP project is here too.

The last time the subcommittee had a briefing on NGNP was in April of 2011. Today the members from INL will update us on the licensing framework, our development work that has taken place between the NGNP project and the NRC staff. And I will present an update of the NGNP fuel research and development work as well.

Members Corradini, Rempe, and Ray have some potential organizational conflict, hence they will not take part in any discussion specifically related to their work.

The rules for participation in today's meeting were announced in the Federal Register on December 17th, 2012, for an open and partially closed meeting, if necessary. However, I understand that it will be mostly open meeting today.

In case we need to discuss any non-pulic information, I am asking INL to identify the need for closing the meeting before we enter into such discussions.

We have a telephone bridge line for public and stakeholders to hear the deliberations. To minimize disturbance, the line will be kept in a listen in only mode until the end of the meeting, when we will provide an opportunity for any member of the public attending this meeting, and person through the bridge line, to make a statement or provide comments.

As a transcript of the meeting is being kept, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the subcommittee. Participants should first identify themselves and speak with sufficient clarity and volume to be readily heard.

I also want to mention, we have a really tight schedule today, and a lot of material to go over. We have to stop earlier than normal because of a separate meeting that was scheduled for later this afternoon. So we have to finish by 3:00.

And we have a short lunch break. Some of us have to run off to another short meeting at that time. But we'll be back in time to keep the meeting going.

We're open for questions as usual, but we really need to give them as much time as we can to get through the presentations. We'll now proceed with the meeting. And I call upon, well, actually, are we going to start with Don, or --

MR. CARLSON: Yes.

CHAIR BLEY: Yes, I call upon Don Carlson of NRO to introduce the meeting.

MR. CARLSON: Thank you. Good morning, I'm Don Carlson. I'm the lead project manager for the NGNP project in the NRC Office of New Reactors, Division of Advanced Reactors and Rulemaking.

We've been engaged in some discussions and interactions, white paper reviews, et cetera, on these high priority licensing and policy issues for NGNP, or modular HTGRs, for several years now.

Our plan is to finalize some feedback to DOE and INL on some of these issues, and present that to the subcommittee in a few months. So the purpose of today's briefing, it's an information briefing, and it's between the ACRS members and DOE/INL.

There are, of course, some NRC staff in attendance. But they are to participate as observers only, and so I would remind the NRC staff of that, to keep the discussion between DOE/INL and the members.

MEMBER CORRADINI: Can I ask a question then?

CHAIR BLEY: Certainly.

MEMBER CORRADINI: So the product of this is exactly what? Because since NGNP is in a genericizing mode these days, what is the staff going to present to the ACRS that we have to comment on, at the end?

MR. CARLSON: Well, DOE/INL has asked us to provide feedback on a number of issues. And they had actually provided the NRC with reimbursable funds to pursue that.

And the four big issues we've been talking about off and on for modular HTGRs and advanced reactors in general for many, many years now, since the 80s, so it's licensing basis event selection, source terms, containment function and performance, and emergency preparedness and planning.

MEMBER CORRADINI: Okay. And just so I'm clear, so I don't know what form this will take. This won't be an SER. So it'll be a NUREG from the staff? What form will it take and what sort of response are you expecting from the ACRS.

Because to me, if this is kind of in a wrap-up mode, I want to make sure there's a clean cut so future people know what to pick up and work on.

MR. CARLSON: It will be less formal than the NUREG. And it will be, as we're now formulating the final feedback, it will be in the form of three documents.

MEMBER CORRADINI: Okay.

MR. CARLSON: Updates to the publicly issued white paper assessment reports that were issued about February last year, and a new document that summarizes our feedback on those issues under the four headings I just mentioned.

MEMBER CORRADINI: And then refers back to the assessment reports?

MR. CARLSON: Refers somewhat back to the assessment reports for more detailed discussions.

MEMBER CORRADINI: Can I ask one other thing?

MEMBER ARMIJO: When would we hear about

your feedback report? When would we see it?

MR. CARLSON: It would be a month in advance of the staff briefing on these topics, which is now scheduled for April.

MS. BANERJEE: April 9th.

MEMBER CORRADINI: And the assessment reports are post or pre the responses from the RAIs of DOE back to you guys?

MR. CARLSON: What we wrote and issued in February already incorporated the RAI responses. And so what we have done since then is had a series of interactions in the form of public meetings and public conference calls where DOE and INL have responded to the feedback that we provided initially in those assessment reports in February.

And so we've been refining, clarifying, modifying our feedback to them on those topics based on those interactions.

MEMBER SHACK: Are you going to update the assessment reports?

MR. CARLSON: Yes. We are updating the assessment reports and they will be called staff positions. If you looked at the earlier ones, they were called working group positions. And we didn't put that through there. There was intensive concurrence process at that time.

CHAIR BLEY: But they will go through concurrence before we --

MEMBER CORRADINI: They will through concurrence now.

MR. CARLSON: They will.

MS. BANERJEE: And just to remind the members, we have a April 9th subcommittee meeting where staff is going to present their side of the story.

And 30 days before that, they are going to give us a copy of their revised assessment report and position document that they're talking about.

MR. CARLSON: Exactly.

MS. BANERJEE: And then in May, full committee, we have scheduled another briefing for letter writing.

CHAIR BLEY: Okay.

MR. CARLSON: Good. Okay, I'm finished.

CHAIR BLEY: Okay. I'll turn it over to Doctor Carl Sink.

DR SINK: Good morning. Carl Sink, I'm the program manager at DOE for the next generation nuclear plant demonstration project. Much of what I was going to say has already been touched on.

(Laughter)

DR. SINK: So I'll quickly go through the introductory slides that I've prepared, mainly just to recap the process that we've used to get us to where we are today.

The Energy Policy Act of 2005 directed DOE and NRC to work together to put together a licensing strategy for high-temperature gas reactors for the NGNP project.

And in that, it called out these particular issues that were sticky issues that needed to be covered and focused on for this new type of reactor to be licensed.

In that licensing strategy in 2008, which was sent to Congress, it specified that we would focus on adapting existing light water reactor technical licensing requirements for use in establishing NGNP design specific technical requirements.

And we would also use deterministic engineering judgement and analysis complemented by probabilistic risk assessment in doing that.

The Nuclear Energy Advisory Committee in 2010 and 2011 reviewed the status of the NGNP project. As part of that, they looked at the status of our regulatory development and, in their final report to the Secretary, recommended that we continue our interactions with the Nuclear Regulatory Commission to develop the licensing framework.

And when the Secretary of Energy forwarded that report to Congress in October of 2011 he specifically endorsed the need to continue this work with the NRC.

I believe our last briefing to the ACRS from DOE was in 2008, just after the licensing strategy document was produced. And since then, we've been undergoing the process that Don briefly described where we prepared white papers, which summarized how the existing light water reactor requirements would need to be adapted for the NGNP.

And these white papers, and their submittal dates, are listed here on the next three slides. And it also shows, in the right column, public meetings that we held for interactions between the NRC and DOE throughout the past three years, specifically.

MEMBER REMPE: Carl, I was looking ahead at some of the presentations and they're showing a prismatic design. Has the decision, prismatic versus pebble, been made yet?

MR. CARLSON: No. There's no official

position by DOE between prismatic or pebble.

MEMBER REMPE: Okay. And also, what's going on? I saw the alliance wanted to do something now in Georgia they're talking about maybe building. Is that anything significant, or it's still just very --

MR. CARLSON: What you may be referring to, the NGNP Industry Alliance has recently gotten a new member. And it's the Savannah River Community Reuse Association has joined their membership.

And so just once again, the alliance is reaching out to industry communities and others to find out what their options would be for using NGNP in America.

MEMBER CORRADINI: Carl, can I ask another question then?

MR. CARLSON: Sure.

MEMBER CORRADINI: For the last report, INL X1122708, the safety basis, that's an accumulation of all the other reports, as I gather it, to essentially roll up to what's the safety basis for a design, no?

MR. CARLSON: Actually, no. That was a report specific to modular reactor safety base. It was submitted for information only. And from the staff, it was requested that we not have any public meetings on that yet.

MEMBER CORRADINI: So is that technology neutral supposedly, no? That's prismatic or pebble either.

MR. CARLSON: It's either. But it's focused on the modular aspects of --

MEMBER CORRADINI: Oh, the modular aspects.

MR. CARLSON: The modular aspects --

MEMBER ARMIJO: Excuse me, multiple.

DR. SINK: Multiple modules that are --

MEMBER CORRADINI: Okay. I'm sorry. All right, I understand. Sorry about that.

MR. KINSEY: Excuse me, this is Jim Kinsey from the INL, just another point of clarification, Dr. Corradini.

The other piece of dialogue that we had with the NRC staff on the safety basis document is that, as these staff positions were being developed and that material was going to be routed through the staff and the NRC staff's management, it was felt that it would be handy to have a 30 to 40 page summary document that summarized a lot of the material in the white papers, and also the aspects of modular HTGR.

So that was really its purpose. It wasn't for feedback again. It was to provide sort of a handy set of notes that described the design.

MEMBER CORRADINI: So then it is kind of a summary? The way you just discussed it, it kind of of summarized the --

MR. KINSEY: It's not a summary of the positions that we've proposed. It's more of a summary of modular HTGRs and their safety aspects.

MEMBER CORRADINI: Oh, fine. Thank you.

CHAIR BLEY: Well, do we have that white paper, ACRS?

MEMBER CORRADINI: I don't think so. Is it in the CD you sent us, Maitri?

MS. BANERJEE: Yes, 22708 is part of the CD.

MEMBER ARMIJO: Yes. We can look at that.

MEMBER CORRADINI: Thank you.

DR. SINK: Slide 8, okay. So as part of our interactions with the NRC, we have received approximately 450 requests for additional information that we have gone through. And we responded to those.

Most of that work was done prior to 2012, when we received the assessment reports from the NRC staff that are shown, for fuel qualification, mechanistic source terms, defense in depth, licensing event selection, safety classification of systems, structures, and components.

So starting in February, going onto the next slide, in the spring of 2012, trying to focus in and wrap up this process, bring it to some sort of conclusion, there was a dialogue between the NRC staff and DOE to focus on these four key areas that had been already discussed in public meetings.

We began to have public meetings in the spring of 2012 on that. And, as a way to take another step forward toward bringing the process to closure, NGNP transmitted a letter to the NRC in July, which summed up our positions and our requests for a staff position on these topics. And so those are the four key areas listed there, in the key.

So today we will be presenting on our topics, which support the licensing framework, which have been developed. We're going to give you the technical and the regulatory background for the positions that DOE has come up with.

I just want to restate that DOE is

focused on the resolution of some long standing topics that have been around for a long time. There's a lot of history here.

And we're trying to move some of the cloud of uncertainty that has been around some of these topics for quite some time.

These are topics that are raised with us from the private sector, that they're wanting to have clarity on, so that they could submit their documentation to the NGNP, and for us to continue working on necessary R and D to support that.

MEMBER CORRADINI: So can I ask a question? It's a bit off topic? So these topics are generic. So would they influence any other sort of advanced reactor technology that the NRC might consider?

DR. SINK: We believe it does. We've had a lot of feedback from the SMR community that the process that we've used, and some of the topics that we've touched on, enlightened their process for what they're going to be doing.

MEMBER CORRADINI: Since you guys have just awarded the SMR, I don't remember the right title for it, but to the B and W, mPower design, is it expected that mPower is going to use some of these analyses and discussions as a basis for their discussions with the staff?

DR. SINK: I don't know so much so far as the analyses. I'm not clear on that, in so far as the process.

MEMBER CORRADINI: Okay.

DR. SINK: Any other questions?

MEMBER ARMIJO: Is it your expectation that these four key issues, you'll have a firm staff position on acceptability of your proposals, or none?

DR. SINK: That would be our hope. But some of the anecdotal feedback we've gotten from the staff is that the positions may not be as strong as we had hoped for.

So we have not seen those yet. And so our understanding is that they're in agreement with the discussions that we've had. And that we won't be surprised by what we see. But I guess what I've heard is they won't be as strong, maybe, as we might have hoped for.

MEMBER CORRADINI: What do you define as strong?

DR. SINK: Well --

MEMBER CORRADINI: So I'm just trying to understand. If there's a gap, I want to understand what your expectation is versus what we will hear in April.

MR. KINSEY: Excuse me, this is Jim Kinsey from the INL. Probably the best short answer to that question is we had some dialogue with the staff in order to clarify expectations.

And we sent the letter in early July. I think it was July 6th. It kind of gives a punch list of the specific items we were looking for their specific feedback on.

MEMBER CORRADINI: Do we have that in the CD? I didn't see that.

MS. BANERJEE: Which one? I'm sorry.

MR. KINSEY: It's a letter from DOE to NRC, July 6th of 2012. I believe it's on there.

MS. BANERJEE: Yes. July 6th letter is in there.

MEMBER CORRADINI: Okay, thank you.

MEMBER REMPE: Here, this letter.

MR. KINSEY: I think we summarized a lot of its scope in the various slide sets here.

DR. SINK: Yes, many of the presentations you received today specifically call out what the request for a staff position was.

DR. KRESS: Which of these five issues do the design basis accidents fall under?

MEMBER CORRADINI: The LBEs.

DR. SINK: The licensing basis event selection process, the second. If there're no further questions, move on to the first presentation by Fred on safety approach and design basis.

MR. SILADY: Good morning. My name is Fred Silady, technology insight supporting the DOE/INL licensing effort. The purpose of my presentation this morning is to briefly provide a summary of the safety approach and design basis.

Many of the topics will be delved into in later presentations in more depth. And a lot of these things many of you have heard over the years, over the decades. And so it's a normalization kind of presentation, more than anything else.

We can skip this slide. I think everybody knows the agenda. The design objective has been pretty constant since the MHTGR pre-application interactions in the late 80s.

Qualitatively, we want to build a reactor, and operate it, that does not disturb the normal day to day activities of the public. And I said reactor. I really should have said plant, that has multiple reactors.

And we were the original SMR, I guess you might say. And to put that into quantitative terms, that means meeting the EPA's Protective Action Guidelines at the plant boundary, not out at some EPZ ten or more miles away, down to very low frequencies on a per-plant-year basis.

Next slide please. These things you know well. You know that we chose three separate entities, the coolant, the fuel, and the moderator. And they each do their job and they're all compatible chemically.

And the characteristics are listed there in terms of the helium coolant. It's neutronically transparent. It's inert chemically, has a low heat capacity, and it's single-phase.

The ceramic coated fuel has a high temperature capability, and very high radionuclide retention. The graphite moderator, separate from the coolant, is high temperature stability, large heat capacity, which results in the long response times.

We took those three things and we developed a simple modular reactor design with passive safety. We decided that the best approach was to retain the radionuclides within the fuel at their source.

We configured and sized the reactor for passive core heat removal from an uninsulated reactor vessel out radially to an external passive cooling system.

This passive heat removal is completely passive. And it'll work whether there's forced or natural circulation of the pressurized or depressurized helium within the primary boundary.

We have a very large negative temperature coefficient that's been demonstrated at several of the seven HTGRs that have been built to date around the world. There's an eighth now being developed in China.

There's no reliance on AC power. There's no reliance on operator action. And it's insensitive to incorrect operator actions.

Next slide. So these are our multiple barriers to radionuclide release. The kernel can retain many of the radionuclides in and of itself. There's multiple coatings, of which silicon carbide is the most important. And the coatings are the most important barrier in this whole list of five things.

The particles are very small. You've seen them. I forgot to bring my little show and tell hand out to pass around. They're compacted into a matrix in graphite, within the fuel element, either form, either as a pebble or block. So that composes a fuel element, and there's pictures to come.

There's a helium pressure boundary, three vessels, more discussion on that to come, and a reactor building. You'll see it here soon.

Next page. So in the upper left hand, in the middle, is the fuel kernel. And then on top, around it, are the various layers. The silicon carbide is the key one.

Now, those are then, and you see them by the pencil point there, those are the particles. This shows the prismatic, how the particles are compacted into almost like a lipstick-size compact.

Multiples of those, 10, 15 of those are put into fuel elements that are quite large. You see it next to a chair there. They're 31 inches high, 14 inches across the flats. So that's the fuel element and the three barriers in the multiple barrier functional containment system.

Let's go to the next page. The pressure boundary is the next barrier. It completely encloses the reactor core, which is made up of those fuel elements. It's made to the Section III vessel standards.

Higher pressure cold helium is always in contact with the vessels. Also the helium pressure does not cause loss of cooling. And this is a big difference, and sometimes we just tend to overlook it and slip into existing reactor thinking.

But all the seven reactors, Fort St. Vrain for instance, if you lost cooling in terms of if you lost pressure of helium, you could continue with the circulators to cool the core.

Next page. This shows one reference, MHTGR, from the extensive interactions we had with the NRC and the ACRS in the 80s.

This is the reactor building below grade. It includes the three vessels. Again, it encloses, so these are nested barriers. It completely encloses the three vessels, the reactor vessel, the cross vessel, and the stem generator vessel.

And its main function is to provide structural protection for that vessel system, whose main purpose is to provide maintenance of core geometry.

So you see some of the characteristics of it, it's seismic grade, it's very thick. The silo part, the cylindrical part that's all below grade has ground surrounding it.

And it has a leak rate that is above what existing reactors have. However it is vented, which has a very important purpose for a noncondensable helium coolant.

Next page. This design was formulated at about the same time that the original advanced reactor policy came out. Fred Bernthal moved that through the Commission.

And it reads like a spec for the MHTGR, to use inherent or passive means of reactor shutdown and heat removal, long time constants, simplified safety systems which reduce required operator actions -- and we're looking to design it so it doesn't require any at all, much less reduce -- minimize the potential for severe accidents and their consequences, safety system independence, incorporate defense in depth, citation of existing technology, or which can be established by commitment to a suitable technology development program. That's on the discussion of where we stand on that. It's on the agenda later today.

Next page. So a key element of the safety philosophy is retain the radionuclides at their source. That requires a lot of effort up front.

The manufacturing process must lead to high quality fuel. Normal operation performance must limit the potential for any radionuclide release during off-normal conditions. So we monitor the coolant in real time.

Then if you have an off-normal event, given that you made it correctly, and given that you've operated it correctly, and you can show that with that monitoring, then we just need to only limit the potential for the delayed radionuclide release, which comes out as we heat the core up passively to get the heat out.

There has to be a gradient, so the core has to go up in temperature. And we sized the reactor long and slender, angular geometry in the design that is receiving the most attention now, such that the release is limited.

The radionuclides are retained at the source because the temperatures are way below the limits that the fuel can take.

MEMBER ARMIJO: With or without forced cooling, even at atmospheric pressure?

MR. SILADY: Yes, that's correct. Next page. So that means some things to the design and to the R and D. It means that this is almost a repeat, same three bullets parallel to those that we talked about before.

We've got to have the manufacturing quality, and the normal operation fuel performance, so that we can stay withing the offsite dose limits. And again, recall from the very first slide, we're trying to meet the Protective Action Guides at the site boundary, the plume exposure, one rem.

And the safety design and technology development focus is on limiting the incremental releases. And the AGR fuel development program has promising results to date.

Next page. Now, this is a functional diagram that, at the top, would apply to any reactor. Keep the people away from the radiation source, retain the radionuclides, the radiation within the core, and the processes, and the spent fuel, and storage, and so on. For the personnel, control the radiation transporting, control the direct shine, or direct radiation.

At this level then, three from the bottom, it begins to be a little specific to the NGNP. Everybody has a control transport from the core. You see a helium pressure boundary in there, which is different.

You see control transport from the reactor building. And we're intentionally not calling it a containment or a confinement, so as to mean different things based on history. And of course you control the transport from the site.

Now as we go down lower, this is where it is HTGR specific, all HTGRs though, modular HTGRs, Control radionuclides in the fuel particles, retain radionuclides in the compacts and elements.

And then there are three key things to keeping the radionuclides in the fuel particles. We're going to remove the heat, or control the heat generation, that means reactivity as well as other things, control chemical attack, we've got a helium coolant, but we know we've got water and air that may challenge the reactor internals, the graphite.

So what is shaded is what our objective is. That is that we can show that for design basis events, that lead to design basis accidents in Chapter 15, that we can meet, 10 CFR 50.34, which is the requirement at the site boundary.

Next page. I need to speed it up just a little bit here. What I'm going to do now is take those three functions at the bottom, the passive heat removal, and control heat generation and chemical attack, and use that as a mini outline for the next three or four slides.

I think I've touched on this one already. Probably the only thing that I need to say is the reactor cavity cooling system surrounds the reactor vessel. And it can be either air or water. It's natural convection.

Next page. This shows a plan view. Many of you have seen this as well. This is the prismatic. Each one of those little hexes is one of those big hexagonal blocks that weigh 300 pounds.

You're seeing the top of one layer. they're stacked ten high, so there's a very large array of fuel elements. This is a very low powered entity relative to other existing reactors. And you can see --

MEMBER ARMIJO: Do you have a numerical value of kilowatts per meters?

MR. SILADY: Yes, it's six watts per cc, or less. So I think the LWRs are in the 60 to 100 watts per cc, or megawatts per meter cubed.

MEMBER ARMIJO: Okay.

MR. SILADY: So you can see the factor there, ten or more. Some of the other designs, more recent SMRs, may be lower.

This annular geometry, why go annular? Well, it gets the heat out nearer to the uninsulated reactor vessel. It shortens the conduction path, and enhances the surface-to-volume ratio.

Next slide please. Now, we have independent means of providing forced cooling, one for normal operation to make power, and one for shutdown conditions to be able to maintenance within the helium pressure boundary.

If those fail, and in addition we lose pressure -- either intentionally the operator depressurizes, like for refueling or whatever, or as a result of a leak or break in the helium pressure boundary -- that's what we call a DLOFC, or a depressurized conduction cool down.

And the core then gradually heats up. And I'll show you a transient. And the heat is removed by the heat transfer processes of conduction radiation convection. And it says rapidly to the reactor vessel, or it says radially. And that's the correct word.

There are generally three phases, although there can be some overlap. The depressurization depends on the size of the leak or the break. The core heats up over a period of days.

And the cool down, after it peaks, takes many days as well. Fort St. Vrain, much larger than these modular HTGRs, to get back down to refueling temperatures it took six months, if you didn't do anything. So low power density, gradual heat up, gradual cool down. Next page.

MEMBER CORRADINI: Just for a matter of just comparison, what was the power density for Fort St. Vrain? I forget.

MR. SILADY: It was also --

(Off microphone comments)

MR. SILADY: 6.3.

MEMBER CORRADINI: 6.3, thank you.

MR. SILADY: I was going to say six, so Dave has it right.

MEMBER CORRADINI: Okay, thank you.

MR. SILADY: Next page.

MEMBER SHACK: Can you refresh my memory? What's the size of that cross vessel?

MR. SILADY: Cross vessel, let's see, it's 22 square feet. So we can divide and figure it out. But it's feet. It's a very big vessel. And we'll talk a little more about it when I show you the picture coming up here, of this.

Next page. These are the transients for

that DLOFC. And this is showing the peak sensor. It doesn't always stay in the same spot. It can go anywhere. But it's generally near the inner graphite center that's not fueled.

The average is the average. You can see that it takes those days to come up. And you can see it takes a long time to come back down, 1,000 hours shown there.

So if you lose forced cooling, and if you're depressurized, this is what you get. If you lose forced cooling, the multiple MEANS, and indefinitely I'm talking about, the operator doesn't start it back up, and you're pressurized, the temperatures are lower. Because you have convection. So they go to maybe 1400 C. Next page, yes?

MEMBER CORRADINI: So, this is from the '89 analysis?

MR. SILADY: Yes, yes it is. That's what the MHTGR means. We took those transients.

MEMBER CORRADINI: I was just guessing.

MR. SILADY: Yes, that's correct.

MEMBER CORRADINI: And when they did that calculation, when they say maximum, that's maximum with uncertainty, or maximum on some sort of best estimate set of calculations because --

MR. SILADY: This is best estimate. And you can put an uncertainty band around it. And we did.

MEMBER CORRADINI: Okay.

MEMBER ARMIJO: And was that for this annular fuel load?

MR. SILADY: Yes. Yes, sir, correct. All this is consistent on MHTGR. I didn't want to muddy it with PBMR and other designs.

MEMBER CORRADINI: Thank you.

MR. SILADY: This is MHTGR again. And this is a slice from the inside out. It shows that central reflector. It makes it look really big compared to the active core. But of course we're not looking at R squared going on here. And you see the three rings in that red active core are the hex blocks.

And then we've got some side reflectors there. It's removable, which is the R. It's permanent, doesn't get removed near the reactor vessel and then out to the silo, which has the reactor cavity cooling system all the way around it.

And this runs top to bottom. Well, the message here is that there's only a little bit of the core that gets as high as that peak temperature,

1600. And that's why the average that's shown below in the previous slide was lower.

Next page. Now, this is from the German experience on the fuel, on their testing. And it's detailed, and it's a little difficult to read all the various lines there, and the cross-hatchings.

But what they did over a period of years in the '80s, and this is what the AGR program is now doing with UCO fuel. This is UO2.

And you can see that for one sphere they had about 15,000 particles in it. And so it crossed. With there, it is says level of one particle failure. And up the side it has a fractional release of krypton-85.

So if you went above that Level 1 particle, you knew you had one or more particles that failed. And if you stayed below 1600, you didn't get any particle failures.

This is just the release coming out of the small 10 to the minus 5 fraction of the fuel that has one of its particles degraded, or not coated properly.

And you can see up to 500 hours now, at 1600 constant. And we were never at 1600 in the transient, but maybe 50 hours. And yes, if you go to

1700, maybe you're starting to get a particle failure there, 1800 you did, 2100 you did, hundreds of degree margin to where we're designing it. The rule of thumb is 1600 C. Next page.

DR. KRESS: Does your primary coolant system measurement just look at krypton? Or does it look at --

MR. SILADY: It's primarily krypton. But it can pick up other nuclides as well. This is the best measure. Next page.

Now, I moved from a heat removal to heat generation. They're very large negative temperature coefficient. ABR in Germany, they would shut the reactor down by turning the circulator off.

They didn't bother to put the rods in. Well, they could quickly, with that negative temperature coefficient, shut it down with turning the circulators off first and then put the rods in.

They're two independent diverse systems, reactivity control, there's control rods that go up and down from the top, and there's a reserve shutdown system, a little boronated right circular cylinders that get dropped from hoppers, completely independent. They both drop on loss of power, the control rods do.

Each system is capable of maintaining the reactor subcritical. One system can maintain it cold shutdown during refueling. And this is relied on for off-normal events, such as rod withdrawal. Or water ingress we know has a positive reactivity addition.

Next page. Now, I'm going to talk about chemical attack, one page for air and one page for water. We can get into this now, or later, whenever you want. If there's more questions, I'm not inviting questions, but it seems to be something that people have on their minds.

With regards to air ingress, we start off with a non-reacting helium coolant. We've got high integrity nuclear-grade pressure vessels that make large breaks seem exceedingly unlikely.

And there's a slow oxidation rate, because we have high purity nuclear-grade graphite. If any air were to come in, say after one of those DLOFCs, the core heats up, it comes back down.

When it comes down, you'll get contraction. You'll get some air in if you wait those hundreds of hours on every one of those. It'll be a mixture of helium and air, because the helium went air, pushed some of the air out. But there's a slow oxidation rate. And it's limited by the core flow area and friction losses.

And again, the fuel particles are not next to the coolant. They're in those compacts, they're within the fuel elements, away from the coolant holes, which are in the middle of the core, not at the graphite that is the reflector or supporter.

Reactor building is embedded. It has vents that close and they operate on Delta P. And so after the helium blows down, the vents close. So there's some limitation then of air in the reactor building.

Next page. The water, which we found out is more risk significant than air because we have a steam generator in the steam cycle design that's at several thousand PSI, and we're only operating the helium at 700 to 1,000 PSI.

So if you get a leak in a steam generator tube, the water comes in. And if you do not isolate, we have isolation that does not require AC power, it's DC power.

And we have moisture monitors that would detect it. Because it's a nuisance to clean them up. We know that fully well from Fort St. Vrain. And we have a dump system.

But if the water came in, and you didn't dump it, and you didn't isolate it, and so on, the relief valve will lift. That's the relief valve on the vessel system.

And now you've got a path for that, which is in the circulating activity. And you've got concerns about the water getting to those particles that don't have the silicon carbide. And you'll have some oxidation in the graphite as well, again, a helium water mixture. Next page.

MEMBER REMPE: Before you go on --

MR. SILADY: Yes?

MEMBER REMPE: Moisture monitors, could you talk a little bit about what it is you have, and are they going to be safety related in this design?

MR. SILADY: At this point, we're relying on the steam generator isolation for the safety related. That's what we came to in the MHTGR. We need more design detail to make a real choice on what's going to be safety related.

The moisture monitors, the technology itself, was demonstrated at Fort St. Vrain. I'm not an expert in that area to tell you exactly. Dave, you know from Fort St. Vrain, or anybody in the room?

PARTICIPANT: No.

MR. SILADY: We can look that up for you.

MEMBER SHACK: The moisture monitors wouldn't be the signal for the isolation?

MR. SILADY: No, separate diverse signals. The moisture monitors are in there for investment protection and down time. If those were to fail, we can measure on high pressure. We can measure on water in the unit, because of the neutronics.

There's several independent means besides the moisture monitors. So we've always approached the design in a very methodical systems engineering boring process.

First, you focus on how to make the power, second, how to protect the investment, and third, given that you've done everything right with the safety design approach, you look at what you need to add over and above.

And that's how we got the isolation. The dump system wouldn't be needed either for safety reasons. It's based on MHTGR reasoning.

MEMBER CORRADINI: Maybe this is a design detail, but I forget. So is the pressure on the steam side higher than the --

MR. SILADY: Yes. Yes, sir, by double.

So not only does it have these means to get water in, and oxidation, and so on. But has a transport mechanism, high pressure transport mechanism to take the circulating activity, some of the plate out and so on, out to the reactor building.

So this is the key. It's frequency and consequence make it risk significant, not that we won't meet the PAGS at the boundary with margin. But of the things we have that challenge the PAGs, this is the family. Next page.

MEMBER ARMIJO: Just a quick question on steam generator isolation. How do you do that. There's no big valve that --

MR. SILADY: Oh, it's DC power stored energy that thermal hydraulically closes the valves, if you will.

MEMBER ARMIJO: Okay. And then you can also just dump the steam generator to get rid of water. Is that --

MR. SILADY: Yes. And we would intend to do that, yes. But some of these things we're doing, so as to get back up to power if we clean up the water, and other things we're doing to make sure the public, and the things we do to make sure we get back up to power, certainly help the public as well.

MEMBER ARMIJO: Yes. And the water graphite reaction, what is the dominant chemical reaction that you have?

MR. SILADY: It's water gas, H2 and CO.

MEMBER ARMIJO: Okay. So you would form hydrogen.

MR. SILADY: Yes. And we have to look at that when it gets into the reactor building. And again, there were a lot of RAIs in the MHTGR days on those as well. We don't have any explosivity considerations, or flammability, but maybe it depends on the reactor building design too.

MEMBER ARMIJO: Yes. With that large volume of graphite, you'd have to have an awful lot of water before you start even getting to the fuel.

MR. SILADY: Yes, definitely. And that's the key, retain the radionuclides at the source. Okay, I'm going to keep moving so as not to blow everybody else's schedule.

So I'm in summary mode here. I told you about the objective. We are going to meet the EPA PAGs at the boundary by retaining the radionuclides at the source. We're responsive to the advanced reactor policy.

MEMBER SHACK: Here it says within and

beyond the design basis. Before you were careful to say DBEs. Do you --

MR. SILADY: No. I was careful probably with regard to 10 CFR 50.34. But we want to meet the Protective Action Guides in the design basis and the beyond design basis.

So that's where that 5 times 10 to the minus 7 number comes from, that we'll be talking about in the next presentation. So you see a repeat of the barriers there. You see a repeat of the functions.

And any further questions? I appreciate your time and look forward to more discussion as we go through the day.

CHAIR BLEY: Thanks, Fred.

MR. HOLBROOK: Okay, the next presentation is on the licensing basis event selection process, which is the second technical presentation on today's meeting agenda.

Slide Number 3 just covers the topics that we will address during this particular presentation. I'll be discussing the risk-informed performance based framework and the top level regulatory criteria, so that I can give Fred a break.

And then Fred will come back in and talk some more, and provide some of the detail behind the presentation that he just gave.

On the topics related to licensing basis event categories, frequency, consequence, curve construction, we'll give you some examples from the MHTGR days, give you a little bit more visual idea of what we're talking about as I go through these slides.

Fred will discuss the licensing basis event evaluation structure, and how we do safety classification of SSCs.

As Carl mentioned at the beginning of the presentation, there was a letter sent to the staff on July 6th of 2012. There were several different topics that were addressed in there under four major categories.

And one of the topics, of course, had to do with risk-informed performance-based approach. And there were some sub-bullets in that letter. These are the sub-bullets you see before you on this screen.

We were seeking to reach agreement from the staff on topics such our use of top level regulatory criteria, and the frequency consequence curve construct that we're using in our approach.

We wanted to reach agreement on the frequency ranges, and the use of mean event sequence frequency as part of our process.

We wanted to reach endorsement of per-plant-year method for addressing our risk at multiple reactor module plant sites, agree on various terminologies that we were using in our approach for naming our event categories, reach agreement on some important points related to the cut-off frequencies for the design basis event region, and the beyond design basis event region, and to reach agreement on a process for accounting for uncertainties and how we come up with these events, and also to address our process for classifying our safety equipment SSCs. So we're proposing a process, oh, go ahead, Sam ---

MEMBER ARMIJO: I'm sorry.

MR. HOLBROOK: Let's go back.

MEMBER ARMIJO: These staff positions, would they be independent of whether you had prismatic fuel or pebble fuel?

MR. HOLBROOK: I think --

MEMBER ARMIJO: The request to the --

MR. HOLBROOK: Yes, this --

MEMBER ARMIJO: This would apply independent of which kind of fuel we chose?

MR. HOLBROOK: Yes. Because this process is at a high level, what we're presenting to you today, as far as the licensing event selection process is at a high level.

Again, we're proposing an approach that is technology neutral, and that allows us to take credit for the inherent safety benefits provided by HTGR designs.

It's comprehensive in that we're going to look at a full range of initiating events, and to evaluate the full plant response to those spectrum of events.

And because we're doing so, instead of single failure criteria we'll be considering multiple failures, and the impacts from those multiple failures.

And once these event sequences are determined, each individual event state is analyzed. And then families of events are compared against the safety criteria, or the top level regulatory criteria, for assessment of safety modules.

The next portion of our presentation will deal with the actual framework itself, and the top level regulatory criteria.

At a very high level, the framework needs to answer these kinds of questions. What must be met? What criteria must be met? When must that criteria be met? How we're going to meet them and how well do we have to meet them?

Now, today's presentations that we're giving you on these topics will focus on those first three questions, what, when and how. And we'll also discuss, as part of Fred's portion of this presentation, some information on design basis accidents. So that also will be discussed.

And during today's presentation, of course, we'll be focusing mostly on licensing basis event selection.

MEMBER CORRADINI: The example that's weaved through here, it still goes prismatic.

MR. HOLBROOK: Yes.

MEMBER CORRADINI: When you need a number you're going to go back to that as a --

MR. HOLBROOK: Yes. Right now it's based on the history of the information that we have. Talk a little regulatory criteria, when we scrutinized the regulations, both NRC and EPA regulations, we were looking for regulations that are generic, technology neutral, independent of the plant.

We're looking for things that were

quantified, but not tied directly to a particular technology, such as core damage frequencies with light water reactors.

So we're looking for things that are generic in nature, but yet still quantitative that we could use. And also we're looking for direct statements of consequences to risks to the public, and to the worker.

During this presentation, as already mentioned, we're going to be focusing on public safety. But that's not to the exclusion later on by future applicants that'll also address our other areas beyond just public safety.

These are the top level regulatory criteria that we have selected. The 100 millirem annualized offsite dose limit, TEDE limit, that comes out of 10 CFR 20, this would be for normal operation and anticipated operational events, or in our terminology, anticipated events.

And we'll get into that in a little more detail here in a few slides. Also the 25 rem TEDE limit coming out of 10 50.34, or 52.79, which is evaluated at the EAB for design basis events, off-normal events.

As we've already mentioned, we're taking into consideration the PAGS, the one rem TEDE limit evaluated at the EPZ. That'll be our design limit for the plant, our design goal, I should say.

And we're also taking into account the QHO, so that we can have a overall assessment of the plant risks evaluated relative to the one mile and ten mile limits.

All these will be included. And again, we'll show you in a few slides how those show up on a frequency consequence curve.

MEMBER ARMIJO: Just for clarification, where did the two hour come from, evaluated the site, EAB --

MR. HOLBROOK: It's right out the 10 CFR.

MEMBER ARMIJO: Right out of the regulation?

MR. HOLBROOK: Right out of the regulation. I think that's for the first two hours at the site boundary, and then 30 days as the plume passes by. But that's wording directly out of regulations.

MEMBER ARMIJO: Got it.

MEMBER CORRADINI: So --

MR. HOLBROOK: Yes, sir?

MEMBER CORRADINI: Can we go back?

MR. HOLBROOK: Yes.

MEMBER CORRADINI: So the two hours is, when is zero, when you do the two hour calculation?

MR. HOLBROOK: That would be in the initiating event.

MEMBER CORRADINI: But since everything is delayed, wouldn't the two hours be two hours be after the start of release of the source term? In other words, the potential for the high dose is shifted in time.

So you have to look independent of when actual release is, you have to look for doing maximum, right? I just want to make sure I'm not off base.

MR. SILADY: I think it's from when the release starts.

MR. ALBERTSON: When the release starts, not the initiating event itself.

MR. SILADY: I don't think you translate it to wherever you can find the greatest two hours.

MEMBER CORRADINI: Yes. If I were the regulator I might do that.

MR. SILADY: It depends on the system. It's all right. We're going to --

MEMBER CORRADINI: I understand. I just

wanted to be sure.

(Crosstalk)

MEMBER ARMIJO: I just wondered whether it was arbitrary or whether it depended on the characteristics of the plant.

MR. SHEA: Just to clarify it, it is the worst two hours.

MEMBER CORRADINI: And you are?

MR. SHEA: Jim Shea of the staff.

(Off microphone comments)

MR. HOLBROOK: The use of frequency consequence curve lends itself to this process. And of course it has a frequency access and a consequence access, as you will see.

Event likelihood is implicit in the current regulations. However, in many cases explicit frequencies are not typically stated. And as Fred will show you shortly here with the frequency consequence curve construction, there is some judgements involved in how to lay those out on the frequency consequence curve. So we'll get to that.

Then sequence frequency is used, since it is a frequency to be compared to the doses in the top level regulatory criteria, and also as compared against the QHOs.

We use a MEAN frequency as a best measure of expected outcome. And I should clarify that. When I say event sequence frequency it's not just the initiating event, but it's the full sequence that you would develop through the development of the PRA concept.

Again, MEAN frequency is selected, however we also will calculate confidence bounds on that MEAN frequency. So for instance, in the design basis event region, we would not only look at the MEAN frequency, but we would look at it as 95 percent confidence level in comparison to other regions.

If an event does fall close to a category boundary, we would look at that frequency band. And if it overlaps into another category, then we would also compare the consequences for that particular event to both category limits.

So we're not just looking at MEAN frequency by itself. We're looking at the upper bound and frequency, or the lower bound, depending on which range that you're in, and comparing against all the applicable criteria.

We're expressing these frequencies on a per-plant-year basis. Obviously this is what's most important to the public. They want to know what's the impact on them. They don't care if it's this plant or that plant.

But it also does give us the flexibility to design for either a plant with one reactor module, or however number, say for instance four or eight.

On the consequence side of things dose limits, of course, are associated with the top level regulatory criteria, are plotted on the curve. Again, we're using MEAN values to select where the plotted point shows up on the chart.

However, we also are in the consequence range looking at confidence values as well, especially for the design basis event region. We would compare those upper bound consequences to the criteria.

As was already mentioned by Fred, we're using the PAGs as a design goal. So we'll be designing to meet the PAGs at the EAB to avoid sheltering the public.

The goal is to bring in the LPZ and the EPZ to the same distance, which we judge to be approximately 400 meters. And this will allow colocation of the plant close to a process industry facility that might need process heat.

And finally, on this slide here in this

portion of the presentation, I wanted to give you a depiction of how we see the event process evolving through the design phases.

That blue arrow through the center is showing the different stages of development of the design, preconceptual, conceptual, preliminary, and final.

The boxes up above show you the evolution of the licensing basis events, whereon the left hand side is your starting point, where you have some deterministic choices that are drawn from prior HTGR experience, or expert insights.

And then you see a development. As the design develops you see an evolution of the licensing basis events progress through those boxes at the top. And along the bottom, you see expected inputs that we would apply to this process as we go through these different phases.

By its nature, this will be an iterative process, because we'll be evolving the design as we go along. Early in the initial phase you draw upon the history of prior HTGR licensing events and expert insights to establish a starting point for use by design development and scoping analyses.

As the initial design is developed, you

can see in the second box across the top where we revised that initial list, because we're starting to develop a PRA at this point. We're starting to develop events based on that PRA analysis.

As the design evolves, so do the LBEs. And during preliminary and final design stages, of course, we'll have many opportunities to speak with the staff, receive their input, let them review what we're coming up with as far as our licensing events, and to incorporate their feedback into our process.

So that kind of gives you an overall view of what we see the process looking like from an evolutionary standpoint.

DR. KRESS: Where do you feel like you are now?

MR. HOLBROOK: Right now, well, the design has not been initiated. We've been in preconceptual design phases up until now. And then, depending how things progress forward with the DOE process and interactions with the industry, then at that point we would enter into conceptual design.

DR. KRESS: Do you have your PRA already?

MR. HOLBROOK: Say again?

DR. KRESS: Do you have a PRA yet?

MR. HOLBROOK: No. We've not got to that

point in --

MEMBER ARMIJO: There's been some work, according to your chart, in preliminary work where --

MR. HOLBROOK: Yes.

MEMBER ARMIJO: -- PRA results were incorporated, I guess from the various development programs that DOE's sponsored.

MR. HOLBROOK: Yes, you're correct. The MHTGR got to the end of the conceptual design, and was starting preliminary design.

MEMBER ARMIJO: So you would just basically go back and reconfirm, as you got --

MR. HOLBROOK: Yes. Plus, it's reasonable to expect that there would be some design differences with a new design, because of the time that's transpired since the MHTGR until now.

MEMBER ARMIJO: Well, it doesn't look that different to me, from what was being proposed 20 years ago.

MEMBER CORRADINI: I think their approach is, if I might understand. My impression is your approach is different than the approach that was used to license with the draft SER in '89.

MR. SILADY: Many facets of the approach are --

MEMBER CORRADINI: No, I'm not talking about the design. I'm talking about the process.

MR. SILADY: Yes, but I think the approach, I'll point them out, with the exception of some terminology changes --

MR. HOLBROOK: It's pretty similar.

MR. SILADY: --is very, very similar. Now, it's much different than Fort St. Vrain, much different than the large HTGRs that GA had sold in the 70s. But once we went to the modular HTGR, the approach here --

MEMBER CORRADINI: Let me just make sure I say it right. Maybe I'm saying it wrong. But in '89 you used 10 CFR 50. You did DVAs. You didn't have to deal with severe accidents in this standpoint.

In this way you're approaching it, it would be 10 CRF 52, or 52 prime, or 53, or something. Because the Commission at least instructed the staff to think of this as a lead on a technology neutral framework.

So the process is different. It's going to require a much more detailed design to march through the staff than it would be in '89. I'm not out of place, right?

MR. SILADY: That's true. You're at a higher process than we were thinking in what we're presenting here today. But you're absolutely right.

MEMBER CORRADINI: Okay, all right.

MR. HOLBROOK: Okay. At this point I'm going to turn it back over to Fred. He's going to carry on with licensing basis event categories, and the frequency consequence curve.

MR. SILADY: Okay. I'm going to pick it up and I'm not in as hurried a mode as I was on that normalization safety basis. And I would encourage questions.

And I'll be addressing many of the things that we've already talked about as we go through here in a little more detail.

So the top of the regulatory criteria applied to the full spectrum of normal operation and off-normal events. But we essentially went through and screened everything in those three criteria, generic, technology neutral, direct measures, quantitative.

And so the 10 CFR, and the EPA regulations, and so on, are not self-consistent, as you'll see when we put them on an FC chart. They each have their own specific range. They each have their own accident rule set.

But rather than coming up with something new, we said, well, we're going to take what is state of the regulatory process here and use that. And we finally came to the realization that there were basically four different kinds of licensing basis events.

In the MHTGR days, we thought there were just the top three, and the design basis accident was another kind of fish. But licensing basis events for the NGNP include these four kinds of things.

Anticipated events, and just a year ago we were calling them anticipated operational occurrences and there was a confusion with the staff in terms of their use of that term in Chapter 15 of the deterministic existing reactors, and so we went to anticipated events. And we hope that hasn't been used, and there isn't a conflict there.

MEMBER ARMIJO: But on a frequency basis it's different, isn't it? It's not once in a plant life.

MR. SILADY: I'll get to that too.

MEMBER ARMIJO: Okay, if you could explain that --

MR. SILADY: Yes, if you could shift it

a little bit.

MEMBER ARMIJO: I'd like to understand that.

MR. SILADY: And design basis events using the whole plant, beyond design basis events using the whole plant, I'm going to talk about each of these individually. So I'll leave something to be said for later slides.

And the design basis accidents that are derived from the DBE's, but now you only take into consideration the safety related structure systems and components.

And we didn't put a frequency range there on this summary slide, but it's going to be somewhere in the DBE or below, or below region, for those DBAs. And I'll show a picture of that as well.

CHAIR BLEY: Excuse me, are you going to get into why you added this category?

MR. SILADY: Oh, it's always been there. The design basis accident --

CHAIR BLEY: Okay, I thought you said --

MR. SILADY: -- is what was called the licensing basis event.

CHAIR BLEY: Ah.

MR. SILADY: And the reason we added it

was to try to, in a risk informed approach, relate to this element of the regulatory process that says you only look at safety related response to initiating events, and the sequence in Chapter 15.

CHAIR BLEY: But they are a sub-set of your --

MR. SILADY: Yes. All right, back to the frequency consequence plot. Mark has talked about the ordinate there, that it's event sequenced on a MEAN frequency basis, on a per-plant-year basis.

So this one part applies to if you have an accident in one reactor, or if you have an accident in multiple reactors, or all the reactors.

Similarly, he's talked about the consequences. For illustration purposes here, we're going to plot all these things at the exclusion area boundary. And I'll point out which ones really don't get evaluated there.

DR. KRESS: When you apply this to more than one reactor, you just multiply the frequency by the number of reactors?

MR. SILADY: Generally it's on a per-plant-year basis. So if it's one or more, that's the frequency. And then the consequences is where the multiple source terms come in. That's how it's handled.

DR. KRESS: I see.

MEMBER CORRADINI: I guess I don't understand what you just said, I'm sorry.

MR. SILADY: Okay.

MEMBER CORRADINI: So when you get to the right point, can you repeat that. Because Tom understood it, but I don't.

MR. SILADY: Okay. Well, let's talk about it now.

MEMBER CORRADINI: He, of course, has been thinking about this for --

DR. KRESS: I didn't understand it either.

MR. SILADY: Okay. Well, let's go for understanding. The event can have one or more reactors involved. If it has any of them involved, it gets plotted at whatever that frequency is.

Now, it may turn out that the event has more than one. And then its frequency may be the same, higher, or lower. Whatever that frequency is, that's where you plot it on the frequency.

DR. KRESS: You're talking about seismic events from this?

MR. SILADY: Yes, seismic would be a good one, loss of offsite power would be another one that affects all four.

DR. KRESS: I see.

MR. SILADY: Or there could be subtle differences. And they're all maintained by the same --

MEMBER CORRADINI: I get it.

MR. SILADY: You got it. Now, when you go to do the consequences though, now, if you've got more than one, that's where you take into account that the release is different. Have we got understanding now? Okay, let's go to that. Oh wait, I have points to make on this.

DR. KRESS: Now, my point was if I have a frequency consequence curve, and I have more than one module on the site, I think I need to multiply the frequency itself by the number of modules?

MR. SILADY: We don't have understanding.

DR. KRESS: I don't think you do that.

MR. SILADY: No, we don't. We take into account what the frequency of the event is. And the event says whether it had one, or two, or three, or four, or whatever.

And then we plot what that frequency is for that one, two, three, or four on there. And sometimes it's multiplication. A lot of times it's not. So we leave that to be figured out in terms of the event frequency.

CHAIR BLEY: But I think Tom was getting at the point, if I have something simple that's not coupled, and I have four modules, then the frequency of that is four times what it would be --

DR. KRESS: Yes, that's exactly --

CHAIR BLEY: -- if he only had one module, if they're completely uncoupled and independent.

DR. KRESS: Probably it doesn't matter, because four times one of these things is not much different than one time.

MEMBER ARMIJO: Still, conceptually you've got --

DR. KRESS: But the principle is there, yes.

MR. SILADY: Four independent events. But you do not want to have the axis have that built in. So it's per-plant-year. And when it is an event that affects all of them, then it has a different frequency, okay.

So you don't do the multiplication. And that's what I'm fighting against here, kind of coming back a little bit. So I see your point. But I want to be more general.

DR. KRESS: Well, it could be something that moves you from one with those areas down to the other one, which you don't want that to happen.

MR. SILADY: If you have ten reactors, or eight reactors, it makes a difference how you formulate this.

DR. KRESS: A factor of ten.

MR. SILADY: Yes, that's right. Now, in the anticipated events, these are things that are anticipated, that in the plant lifetime you can expect them to occur. They may not, or they may occur more than once. But you can expect them.

And in the MHTGR days, we designed 40 year lifetime. Now, these days, the NGNP is 60 year. So rather than seeing this thing move, and having a weird number that wasn't round, we decided to go to 10 to the minus 2 for the anticipated events.

It's not going to affect the design, because of other things. That little factor of 1.6 being off is not going to affect anything. It's more conservative to have this tighter limit that's below background, go down to 10 to the minus 2, than have it be higher. So we believe that it's appropriate rounding, if you will.

Now, the other thing about the anticipated events, the dose criteria is 10 CFR 20. It's not really per event. And we want to use this chart per event, for the most part.

There are two exceptions. And this is the first one. It's an annual basis, 100 millirem. And also that's including normal operation. So that's why you get a kink in the curve.

And if you go back to the MHTGR days, we didn't have that kink there. And so we've gotten a little smarter. If you had ten of these events in a year, with the top point there, on average, they all, if you were going to exactly meet it, would have to be one tenth.

You couldn't have those above one, have the line go straight up. Similarly, you can't have it go diagonally, all the way down to the 10 CFR 20 either.

DR. KRESS: Why could you not?

MR. SILADY: Well, it would be conservative. But the requirement is that for events that aren't expected once in the life of the plant, you never can exceed 10 CFR 20, to draw it tighter.

DR. KRESS: You could have a conservative line.

MR. SILADY: Sure.

DR. KRESS: But you didn't consider that?

MR. SILADY: No, what we're doing is trying to put frequencies on these generally, consequence limits out of the regulations as best we understand them.

Other points here is what are we going to use these events for? Well, they're used for the design of the plant. They're used for possibly tech specs.

Now, how are we going to find what the events are, the ones that are closest to the blue line, the acceptable and unacceptable division? No, they're any event, even have a zero dose, any event that would be outside.

It would be in the unacceptable range if it were not for some function that is being performed to keep them in the acceptable range. There're some SSC that's in that design, intentionally, but maybe unintentionally, there's something in that design that's keeping it acceptable.

We have to know what that is so that we know that it has the right capability, and the right reliability, and so on. So events that are close to the line are important. But events with zero dose are important as well.

DR. KRESS: That's how you arrived at safety related SSC?

MR. SILADY: That's coming next. This region is compared typically in Chapter 11 against the normal operation and anticipated events, using the new term, against 10 CFR 20, 100 millirem.

And we all know that average background natural is 300 millirem. And you have another 300 from man-made causes. So it's a very tight limit that is put on nuclear power plants, incrementally over the background, that people get, which would be 600 millirems, average.

Now, let's go to the next page. Now we go into the design basis events. And we want them to extend from the end of the anticipated events. And the real question is how far do they go down.

We have a per-plant-year axis here. So as we were discussing before, if it is an individual event affecting only one reactor and you had four reactors, in essence we're putting in a requirement per reactor that it be at 2.5 times 10 to the minus 5. And if you had more modules, it would be even lower.

Per-plant-year is the right way to think about this entire construct, in that the public needs to be protected. And they don't care whether there was one, two, or five. They may not even know how many are there. You have to look at the integrated risk coming from the site.

Now, you see that we have a slanted line in this area as well. And I guess I first should go through the frequency side. These are events that are not expected in the plant lifetime.

They would be in a fleet of plants. Let's say that we had a commercial industry on modular HTGRs, and we had 200 plants out there. And they all had four reactors.

It'd be something like less than one percent at 10 the minus 4 that you would have anything in this region occur in the lifetime of the fleet of plants. Now, the consequences --

MEMBER REMPE: So if I'm a designer, and I build a plant to meet this per-plant requirement --

MR. SILADY: Yes?

MEMBER REMPE: And I decide to put four modules in. And later the utility, or the owner/operator, I guess is what Harold calls it, decides to build four more modules. Does that mean the requirement on my first four modules needs to be a bit tighter, because suddenly you've got eight modules that have got to meet this.

And so even though you've licensed these guys for the first four, you're going to have to go back through and say, well, even though it was okay yesterday, today you're going to have to be tighter.

MR. SILADY: That's correct. But let me add a few qualifiers to it. Nobody's going to design right up to the limit. And in fact, our design goal is to design to the PAG, which is one rem, so already a factor of 10 or 25 away from the limit.

MEMBER REMPE: And I'm thinking about this as a technology neutral approach that could be used for a lot of different designs, a sodium reactor design, for example.

If I were doing something like this, then you need to start thinking about the maximum number of plant units, or modules you're going to put at the site, if you're going to do this.

MR. SILADY: You're absolutely right, keep coming. I understand what you're saying. So first response is that, at least for the HTGR, we're going to try to meet the PAG.

But let's say some other technology

doesn't try to do that. They're smart. They know that they need to budget. They know how big their site is. And they know their margins.

I don't expect that any designer is going to design right up to the end of the line. He's going to leave a little leeway.

MEMBER REMPE: It just needs to be explained as part of the approach.

MR. SILADY: And when the time comes, just like there are power upratings, they'll take another look at it, and say where am I relative to that line. And the case will be made, or it won't be made, that they stay within the regulation.

CHAIR BLEY: Fred, the argument you made in the beginning, why this should be done on a plant basis because the public doesn't really care what's inside of that plant, seems to automatically extend to a site, as well as a plant.

MR. SILADY: That's correct.

CHAIR BLEY: Okay, that's your thinking.

MR. SILADY: That's correct. And we decided not to broach that topic and get into that now. We'd like to get an industry going. But if you were to put one of these on a brown field site, already has an existing LWR, what margin is left for us?

CHAIR BLEY: Exactly right.

MR. SILADY: Okay. It's something to think about. All right, we have a slanted line here on the consequence, in that we first proposed it back in the MHTGR days as just going straight up.

And then our reviewers back then said, no, we would never accept for events that are at the top of that region to have 100 percent 10 CFR. Back then it was 100, I guess. And they said ten percent or something like that would be more reasonable. And so we adopted that.

DR. KRESS: Did you consider using K times C equal to the constant?

MR. SILADY: An isotherm, a risk?

DR. KRESS: No risk, in reverse.

MR. SILADY: Yes. There's all kinds of ways.

DR. KRESS: Yes, there's lot of ways you could do that.

MR. SILADY: Right. We are trying to meet the regulations the way they're written. They don't say 10 CFR 50.34 at this frequency should be here, and at this frequency should be here.

When we got the feedback from the NRC at that time, we're proposing this. If somebody thinks it should be 5 percent, or 25 percent, fine, if we think it's got a sound basis that'll last for the duration of the project.

All right, let's go to the next page. Now we add the beyond design basis events. And they obviously have to start at the bottom of the design basis events. And the question is how far down should they go.

We've been using PRA on HTGRs since 1975. And we know there are limitations. And we understand that the ability to risk informed at very low frequencies is limited. You can't assure completeness.

In addition, we looked at the prompt quantitative health objective and saw that the individual risk there was 5 times 10 to the minus 7. And so when we tried to plot that here, and it's being plotted at the EAB, and it really should be at one mile, we came to the realization that there is no dose limit if you're below 5 times 10 to the minus 7.

It's prompt depth. So the line goes flat at 5 times 10 to the minus 7. We could find no other hint in the regulations where the de minimis should be. It has to be there somewhere.

It can't be at 10 to the minus 12th. It can't be at 10 to the minus 5th. We know we have to have the region extend down, and we need to compare to the NRC safety goals.

When we found that 5 times 10 to the minus 7, it wasn't really the roundest in number, but it sure beat 1.6 times 10 to the minus 2. So we said we have something written down that we can use as a de minimis.

And again, we're proposing. And if somebody wants this to go to 2 times 10 to the minus 7, or 10 to the minus 7, fine. If the axis goes 10 to the minus 8, we do our PRAs down to 10 to the minus 8, and try to be as complete as possible.

But there are lots of things that can happen in this world that you can't pick up in a PRA when you get down in that space.

And so we, by meeting the safety goals down to 5 times 10 to the minus 7, which is the top of the regulatory criteria that applies to this region -- it of course applies to all the regions, but this is where it really gets tested -- so that's the basis for the bottom of the region.

DR. KRESS: When I first saw the curve I wondered why the slope of that slanting part was different than the other slopes.

MR. SILADY: Well, it's really not a straight line. But on this log-log scale, it goes through what is the 1 percent chance of death, 5 percent, 10 percent, 50 percent. And it's an S-curve. But it gets straightened out here when you plot it this way.

And let's go to the next slide then. We put the protective action guide plume PAG, again measured in TEDE as most of these are, only the prompt QHO is a little different, just to show where our design goal is. And this gets back to the margins that we have.

And we're going to try to meet that PAG on a MEAN basis. And I'll talk more about the accident rule sets for each of these top level regulatory criteria here in a few slides.

So this is just to keep some perspective of our design goal, relative to the top level regulatory criteria. Next page. Now, I'm going to move into some examples.

MEMBER CORRADINI: Sir, I'm sorry.

MR. SILADY: Sure.

MEMBER CORRADINI: If you could just repeat, the dashed line you're going to meet how again, I'm sorry?

MR. SILADY: That would normally be met at ten miles for existing --

MEMBER CORRADINI: I understand where you want to meet it. I'm trying to understand from a --

MR. SILADY: Analysis point of view, how, the MEAN versus 95th or --

MEMBER CORRADINI: Yes.

MR. SILADY: Okay. We're going to meet it on a MEAN basis, because emergency planning has to be on your best knowledge of what the consequences are. And when you do the MEAN, that's the expected value, taking into account the uncertainties, and the tails, and so that's the basis.

All right, now, I'm moving into some examples from the MHTGR. And I have scads of back ups here. We've got two big volumes of RAIs on the MHTGR. And they've got mini event trees in them. And they've got all the DBEs, and they've got all the AOOs listed.

All I've tried to do here is, skipping ahead past the safety classification topic a little bit, assume that I know what's safety related. And I'll talk about how we do that later.

And I take the design basis accidents,

the design basis events, the ones that have the entire plant responding, that are from that plot, and I take the three dominant ones. They happen to be 6, 10, and 11.

And now I assume that I only have the safety related SSEs responding, so one at a time. The first one's the highest risk one, a combination of its consequences and frequency.

It has an offset tube rupture in the steam generator. The steam generator isolation, safety related successfully, isolates the steam generator. So it's got water in the steam generator that continues to come in through that tube rupture.

In addition, the main loop's out of service now. And the shutdown cooling system doesn't come on. It's not safety related.

And it results in an early and a delayed release from the helium pressure boundary, via the vessel system relief valve opening with that water ingress, adding pressure. That's one accident. I'm going to show you where that is in a minute on a plot.

The next one is that vessel system relief line having a breach. This is about a 13 square inch line. And so now you get, in addition, an immediate and definite loss of forced cooling in this particular accident.

Those things might be available to cool the reactor down. But they're assumed to not be available for a design basis accident. And so the core heats up and you get a delayed release after that initial release from that breach in the helium pressure boundary.

That's the least of the three, in terms of its consequences, and its risk. Design Basis Accident 11 is a smaller leak. It's just an instrument line somewhere in the helium pressure boundary.

And it has the same assumptions that would go on to a passive heat removal, so the core heats up. And it continues to leak for many hours, a day or so, depending upon the size.

And that's actually worse than the big one, because it provides a driving force from the big break of the other line. It provides a driving force when the core is heating up and the radionuclides are coming out of the fuel.

If you don't have a driving force, if it blows down right away, you don't have as great a means to get the radionuclides out.

Next page now. I'm going to show you these points. Okay, let's stand back. This is an old chart. It's got different terminology. It's MHTGR. We used to call the beyond design basis events back then emergency planning basis events, that EPBE.

MEMBER CORRADINI: So to make sure I understand. This is not from any of your calculations. This is the '89 document.

MR. SILADY: That's correct. We're using the --

MEMBER CORRADINI: The green is what you're going to talk about?

MR. SILADY: The green is from the '89 documents as well. And what it is doing is showing you where the design basis accidents are in frequency and consequence space.

Recall that we plot all our points from a risk assessment on this chart, with their uncertainties, right, and left, and up, and down. We then have the topography of the risk for the plant design, for four reactor MHTGR, 350 megawatts.

We back up, and in just a couple of slides I'm going to show you the process, or how we determine what's safety related. But given that we know what's safety related, we then go back to DBE-6, 10, and 11, which are up in the DBE space as black. And we say now we're only going to have the safety related SSCs respond. Those are the green points.

MEMBER CORRADINI: So you assumed what? That's the part I'm missing. You did what again between black and green?

MR. SILADY: Okay. In the green, we say that if it's not safety related, it's not available.

MEMBER CORRADINI: Oh, okay.

MEMBER REMPE: So the moisture --

MR. SILADY: Monitors what was available. It is not available.

MEMBER REMPE: Were not available for DBA.

MR. SILADY: Right. And the shutdown cooling system, which can have forced cooling, is not available. That's correct.

CHAIR BLEY: So the frequency goes up, but because it's in the design basis, the release goes to nothing?

MR. SILADY: The frequency stays the same, depending if the design basis event happened to only have safety related equipment. Or it goes down if it had other things like the moisture monitors, or the shutdown cooling system coming on.

So we have green ones that are identical in frequency, and we have green ones that are lower in frequency. And the consequences vary as well.

If it had shutdown cooling, it didn't have any delayed release. It didn't go up through the transient, and down, and have a delayed release. So we have a movement of the points, when we go to put them into Chapter 15.

MEMBER CORRADINI: So can I just take one so I get it right?

MR. SILADY: Okay.

MEMBER CORRADINI: So DBE-6 becomes DBA-6. And the frequency goes down, but the consequence goes up.

MR. SILADY: And here's DBE-6.

MEMBER CORRADINI: That's eight.

MR. SILADY: No, it is --

MEMBER REMPE: Go up a bit.

MR. SILADY: Oh, DBE-6. Thank you.

MEMBER CORRADINI: No problem, all right.

MR. SILADY: So, it had very low release. The relief valve did not lift. It had zero release. The moisture monitors picked it up, we isolated. That's the definition of DBE-6.

Now, my moisture monitors fail. I don't pick it up as quickly. I isolate, but water continues to come in. And it lifts the relief valve. The consequences go over to here.

MEMBER CORRADINI: Okay, that's what I just wanted to make sure I understood.

MR. SILADY: You understood it correctly. I just had the wrong point.

MR. ALBERTSON: And the frequency goes down, because you don't have all the good things saved in it.

MR. SILADY: Right. The numbering is different because we didn't make this correlation as safety related and non-safety related as we did the PRA. First, we wanted to get the typography. And that is the segue into my discussing the safety related process next.

MEMBER CORRADINI: Thank you.

MR. SILADY: Okay.

MEMBER CORRADINI: I got it.

MR. SILADY: Okay, let's keep going here.

MEMBER CORRADINI: So can I, since you picked three, I can find two. Where did 11 go?

(Off microphone comments)

MR. SILADY: It's down here somewhere.

MEMBER CORRADINI: Okay, fine.

MR. SILADY: Remember that was a real small leak. So there were hours and hours for the operator to intentionally depressurize, or to do other recovery actions. And so the frequency of actually getting to the passive heat removal was very, very low.

MS. BANERJEE: There is another green point on the axis.

MR. SILADY: This one's DBA-10.

MS. BANERJEE: I know, on the --

MR. SILADY: And this one's DBA-6. And these are other DBEs and DBAs that I'm not talking about. They have zero dose.

MEMBER ARMIJO: What is DC-1?

MR. SILADY: This is the event that ended up being the same as DBA-10. They get numbered by the region that they're in. And if they're below 5 times 10 to the minus 7, we keep track of them for the integration of the entire risk, the complementary cumulative distribution function against the QHOs.

But we don't give them an LBE name. Our LBEs stop at the 5 times 10 to the minus 7, unless the frequency straddles, like this EPBE straddled. And so we said, okay, it's close enough that it's going to be an EPBE.

And we have a DBE up here in the AOO region. That's because of its uncertainty band. You can't see it. They get mixed up with the logarithmic upticks.

MEMBER CORRADINI: So since we're noodling --

MR. SILADY: Yes, that's fine.

MEMBER CORRADINI: What's WC-1?

MR. SILADY: Okay. I'll give you the code. I think that's what you're asking. This is a wet conduction cool down. It's not very scientific sounding, that's why I didn't want to tell you.

(Laughter)

MR. SILADY: When we did this in the hallways at GA we said, okay, well, they're wet or they're dry. And if they have a C, they have a conduction cool down. And if they have an F they have forced cooling. And then we had a --

MEMBER CORRADINI: And W means wet, means what?

MR. SILADY: Water, water comes in.

MEMBER CORRADINI: Oh, it's flooded.

MEMBER SHACK: No, it's a steam generator tube leak.

MEMBER CORRADINI: Plus a cool down?

MR. SILADY: Plus a cool down. It's the same as DBE-6, okay. And the dry one, D for dry, is the same as DBA-10. And 11 is too loaded to plot.

MEMBER CORRADINI: Okay, thank you.

MR. SILADY: I'm here. Let's noodle a little bit further. I'm over time too. This Appendix G-2 was an event that the staff asked us about, a failure of the cross vessel. And so I believe that's, anyway, it's an event that the --

(Off microphone comment)

MR. SILADY: Yes. I don't recall exactly what event it was. I think it was multiple modules. I'll look it up. It's another one that the staff found in their review. So it's an example of independent deterministic thinking, saying what if, which we need.

All right, now I'm ready to go. The evaluation structure, I just wanted to have one slide, maybe two slides, on this. Each of these have their own history.

You do certain things certain ways, in some chapter of the SAR, or in some PRA, or whatever. And we had to get this straight in our heads. On the plot, we plot the MEANs with uncertainty bands.

But when we go for an application, we wanted to get agreement that for 10 CFR 20, and 100 millirem, we're going to do those by MEANs at the EAB on a cumulative basis, over all of the events is what I mean.

For 10 CFR 50.34, we wanted that to be upper bound, 95 percent confidence, at the EAB 25 rem. Now, the EAB versus the LPZ, we were making the LPZ and the EAB as well, so that's why we only have EAB.

The two hour versus 30 day, we do them all for 30 days. And we can look at any time frame you want.

The emergency planning EPA PAGs are on a MEAN basis for the reasons we've talked about. You want to have your emergency planning be as accurate as possible.

And the QHOs are the standard complementary cumulative distribution function over the average, from the EAB, one mile out for the prompt, from the EAB, ten miles out for the latent. Next page.

MEMBER ARMIJO: Could you just go back to that?

MR. SILADY: Sure.

MEMBER ARMIJO: Let's assume you meet your design objective --

MR. SILADY: Yes?

MEMBER ARMIJO: -- that the EPZ is equal to the EAB. What's the practical benefit of meeting that?

MR. SILADY: Well, we hope --

MEMBER ARMIJO: From a plant standpoint?

MR. SILADY: We hope that we have more ability to make a strong case that we meet the other requirements with margin. And that when we are in a public forum, talking to the people that might be living around there, that they don't have to have the same degree of sheltering, drills, and so on. Because the EPZ is now at one mile rather than ten.

Now, if you want to go on other reasons for selecting the EPZ, this is the technical portion of the reasons that go into that.

MEMBER ARMIJO: Would you go as far as to say you would not require emergency planning beyond the EPZ, the regulatory --

MR. KINSEY: This is Jim Kinsey, just a point of clarification. There's some material we've provided to the staff. And we've had some dialogue around this point.

A couple of expansions on Fred's answer, if we meet this goal of the EPZ being at the exclusion area boundary, that would tend to influence how you may evaluate the sighting of a high temperature gas cooled reactor in or near an industrial facility where you're trying to provide process heat.

And the other point to this is we would still have an emergency plan. And it would still be applied to an area that's larger than the EAB, most likely, but probably on a more graded approach. And that's the piece that still needs to be worked out.

MEMBER ARMIJO: So there'd still be some sort of a public evacuation, sheltering kind of thing, but not as --

MR. KINSEY: It would be more of an all hazards plan. And again, the details of that need to be worked out. But the point is, some of the more specific controls that typically are applied in an existing EPZ would be applied primarily on the plant site, inside the emergency exclusionary boundary with, again, additional hazards outside of that area being evaluated and developed in an emergency plan that was in more of an all hazards. Does that answer that question?

MEMBER ARMIJO: Yes.

MR. SILADY: Okay, the next slide is a treatment of uncertainties. It's a standard PRA, a standard code practice in terms of how to treat consequence uncertainties. I'd like to just go past that. We're going to talk about these things quite a bit in the mechanistic source term presentation, and so on.

There's one topic in our remaining ten minutes for your questions, and so on, that I really want to talk about. And that's the safety classification.

So maybe one word on the slide there on the PRA, an HTGR PRA is different than an LWR PRA. There's a separate standard. There's intermediate metrics, like CDF and LERF that just aren't applicable.

We're going to do our best to treat all the sources of the radioactive material. We're going to model the systems in terms of maybe a smaller number of systems. But they may not have as much information and experience.

We're going to look at internal and external events, all operating modes, full scope PRA basically. And it's going to, as we've already mentioned multiple times, have multiple reactors. And it's going to look at nearby hazards as well.

Now, let me go to the safety classification. I already alluded to the fact that in order to determine what an AE was, or a DBE was, you had to know what the required safety functions are.

And I showed you that chart, the blue shaded ones, of what we thought the required, it was only down to a certain level. Obviously you have to go lower. You've got to maintain core geometry to remove the heat, and so on.

But we know those required safety functions. If that's a given, then we look at our SSEs. And we say to ourselves, in each of those DBEs, which of the SSEs are available and sufficient to do those functions.

And I'm going to show you an example of that now. I've already touched on that, let's go to the next slide. Let's look at this Challenge B out here in the unacceptable range.

There's something that's keeping this event here, or here, anywhere in the region, in the acceptable range. And we figure out what those required safety functions are.

That's for mitigation. But we can't forget that there could be an event down here that, if it were up here, would be unacceptable as well. We have to prevent these with reliabilities of SSEs. And we've got to mitigate these with capabilities of SSEs.

This is our construct, and we look at both ways. Now, if you don't have events here you don't' have to do this. We haven't found any events over here.

But we have a process that's generic. And we've used this process to look at other reactor types that have events in this range.

Let's go to the next page. I'm going really slow here. This is one of two. But I wanted, before throwing a bunch of yeses and nos at you, to take one event and ask the question, which SSEs are available and sufficient to remove core heat?

This is just one example of one of the functions, one event, one example. DBE-11 is that small leak in that instrument line that we talked about. It's the reactor, the main loop P transport system that produces the steam for the electricity and the energy conversation area available.

In this particular event, there were

higher frequency ones where the answer was yes. But in the DBE range it was not, no. Was the shutdown cooling system --

MEMBER ARMIJO: Could you explain it a little bit more. You've got a small helium leak. Now why aren't these other systems available? If it's just --

MR. SILADY: Because things fail, or the operator inadvertently shuts them off in a response, and he shouldn't have. There's always failure modes. And the success paths are higher anticipated events.

But when they fail they end up in the design basis event region. So this is the path that leads to an event below 10 of the minus 2, and above 10 of the minus 4.

CHAIR BLEY: That's a whole event sequence, not just --

MEMBER CORRADINI: This is just one branch of many branches.

MR. SILADY: Okay, and I could pull out the mini trees and show you it in context with the others.

MR. ALBERTSON: Showing you all the combinations for the particular initiating --

MEMBER ARMIJO: Right, okay. Got it.

MR. SILADY: Okay, so in this particular one, these are nos. It's the design basis events that you look at to lead to whether something should be safety related or not.

And so now the reactor vessel and the RCCS is an alternative set of SSE's that could remove the core heat in the passive way that I showed you.

The reactor has to have the right power to SV and conductivity, and the reactor vessel has to have the right conductivity and the right emissivity. And the RCCS has to convect up above grade, take the heat out.

In this sequence, it is available. We also have the possibility of the reactor vessel, and the reactor building, and the ground around the reactor building, doing it as well.

And we've done a best estimate calculation on that. And it would work if this was a no. We look at all the possibilities. No, no, yes, yes is the pattern for this one. But we don't decide yet.

Let's go to the next page. We have to look across. We wait to do the safety classification until we've got all of our events. We don't go one by one, saying I'll make that safety related, oh, that's active, I'll put two of those in.

This process first gets the foundation supplemented with the deterministic, or the deterministic supplemented with the risk assessment, whichever way you want to look at it, to answer the three basic questions of what can go wrong, what are its chances, and what are its consequences.

Now, it's the same function. And DBE-11 should say no, no, yes, yes. But there are other events in which the shutdown cooling system is available. And that's shown with a yes.

And we stood back and we said, well, it'd be wise to choose one of these as yes all the way across, as being what we're going to rely on in Chapter 15, for all these events.

Should we take the one where it conducts the heat to the reactor building, and the concrete loses its water, and the heat goes to the surface somehow after it goes into the ground?

Or should we take the one that has that reactor cavity cooling system in that we made passively to get the heat out?

And we said, well, we're going to put into Chapter 3 and Chapter 15 the one, the RCCS that we've designed with such reliability and capability, rather than rely on the one that might do it if this fails. So this is the one we selected as safety related. And that's our process.

MEMBER CORRADINI: So can I say it differently?

MR. SILADY: You may. You can help him.

MEMBER CORRADINI: So the red is what you're relying on to set the dot. If all those red yeses turn, one of them, I don't remember exactly, one of these DBEs turn to no you still have, because of just the fact of the existence of the reactor building and conduction, you still have an automatic yes there. But the frequency and the dose would change.

MR. SILADY: That's correct.

MEMBER CORRADINI: And you've done that as a backup.

MR. SILADY: We did it and it fell out that way.

MEMBER CORRADINI: I understand. But I wanted to ask if you'd have that as a backup.

MR. SILADY: If you were to ask me, I have it as a backup. Is it part of the process that I must have a backup, no.

MEMBER CORRADINI: I understand. But you

do automatically.

MR. SILADY: This technology does.

MEMBER CORRADINI: Okay, fine.

MEMBER ARMIJO: But that's inherent in the system design.

MR. SILADY: It's technology-dependent.

CHAIR BLEY: And it's included in the full PRA results.

MR. SILADY: Yes.

DR. KRESS: And that tells you what safety related SSCs you have.

MR. SILADY: Yes.

DR. KRESS: How do you go from there to design basis accident?

MR. SILADY: I go back to each of these events. And I say, even if they have a yes, I'm going to put a no. And I'm only going to have the event have the response of what's safety related, for this function and the other ones.

And I re-run all the design basis events with only the safety related. And those were those green dots that I showed you on the FC chart, just so we knew where they were.

MR. HOLBROOK: We no longer care about the frequency of those events. We just care about the consequences.

MR. SILADY: Yes. It's frequency independent at that point.

MR. HOLBROOK: And you put them in Chapter 15 in the end though.

MR. SILADY: But it gives you an idea of some of the strength of the process, and the conservatism that we end up designing for events way low.

We're not trying to design for way low. We're trying to design up in the design basis region, everything in the DBE. It's not just selected events.

With the PRA we're trying to be comprehensive in the DBE region. But once we do the process, and we go to the Chapter 15 step, we end up having the design for some pretty rare events.

MEMBER ARMIJO: Just to make sure I understand it, these are very high level components, reactor vessel --

MR. SILADY: Yes.

MEMBER ARMIJO: -- RCCS. So everything associated with those components are classified as safety related equipment? Let's say a circulator, is it --

MR. SILADY: There is no circulator in this one, or this one. The circulator is here and the circulator is here. There's no instrumentation in terms of something has to be detected in order to remove the core heat.

It just does it. It's completely passive. Other functions, there's a safety related reactor protection system that says, ah, I detect water. I'm going to isolate, and so on.

But for this function, it's completely passive. So I think I've got it. Now, there's a sub-function --

MEMBER ARMIJO: Just for the integrity of the reactor, and the vessel --

MR. SILADY: You're starting to get to where I was going.

MEMBER ARMIJO: Okay.

MR. SILADY: There's a sub-function to core heat removal. It has to maintain core geometry. So then I get into the reactor vessel supports, I get into the RCCS silo.

And I have to make the reactor building safety related to protect those things. So the process flows down. This is high level, you're right. I'm ready for my summary. But at any point you can stop me.

MR. HOLBROOK: Whatever sub-components that are necessary to support that required safety function would then become safety related.

MEMBER SHACK: But conversely, only the portion of the thing that's needed for the function is safety related. So it's kind of like a 50.69 built in.

MR. SILADY: I'm glad you said that. Because that's correct. And sometimes we have to change our paradigm.

MEMBER SHACK: It's just a proposal.

MR. SILADY: Yes. We have to change our paradigm. The reactor vessel is needed to maintain core geometry. The reactor vessel is needed to remove core heat.

Does it have to keep the helium in? No. So does it have two different functions? Yes. We design it then for the capability and reliability for those functions so that it does those things for those DBEs.

Will it keep the helium in? Absolutely. The owner wants it to, in terms of helium trucks pulling up to the plant.

DR. KRESS: When I think LWRs, they

classify components, and systems, importance measures, Fussell-Vesely, and the others. You don't do that here, because you don't have a CDF in alert. Is that my understanding?

MR. SILADY: That's correct. That's intermediate metrics so they don't have to do a Level 3 all the way out.

DR. KRESS: Yes.

MR. SILADY: And they kind of go through a pinch point. If the core is damaged --

DR. KRESS: It's another lead in, right, design loop.

MR. SILADY: Yes. And we don't have a pitch point. We don't have anything that severe. And we're all over the map. And so, with this technology you've got contributions in water ingress, contributions from helium leaks, and so on, without forced cooling.

I need to get to the summary. All right, Mark put up what we need to meet, when we need to meet it, how we're going to meet it, and how well. And I've talked a little bit about each, but I haven't been comprehensive. I haven't talked about all the things.

But the licensing basis events are the

when part. And we select that early in the design process. Even in the MHTGR back in the conceptual design, we would select them, the examples that we're showing you today. So it informs the design, and informs the licensing process. It's looking in, and it's looking out.

The third bullet, I've talked enough about. I think you understand the safety classification is tied to where the point is, and how it can either be a mitigation or a prevention that we need to do for those. That's my summary. Any other questions or comments?

CHAIR BLEY: Good, thank you very much. We're about to recess only for ten minutes. And we've got to make up the time somewhere, because we do have fixed endpoints.

But this was a really important discussion for all of us, I think. So we'll recess and come back ten minutes from now, 32, that's according to schedule.

(Whereupon, the foregoing matter went off the record at 10:23 a.m. and went back on the record at 10:35 a.m.)

CHAIR BLEY: We're back in session. David Alberstein, please.

MR. ALBERSTEIN: Yes. My name is Dave Alberstein. I work with TechSource supporting Idaho National Laboratory on the NGNP program. And a portion of the meeting here is going to be devoted to functional containment performance and mechanistic source terms. And that will be followed by a presentation on siting source terms, which is to some extent a sub set of the overall mechanistic source term topic.

This is where we are on the agenda, the outline of the presentation, little introduction, little regulatory background. And then get into the details of functional containment performance and how one determines or calculates mechanistic source terms. And then we'll wrap it up with a few conclusions.

We submitted, back in July of 2010, next slide, we submitted a White Paper on the subject of mechanistic source terms. The ADAMS number is there. If we go to the next slide, that paper contained information on radionuclide transport and retention in modular HTGR's, a description of the functional containment, a discussion of behavioral radionuclides in the plant, mechanistic source term models and modeling assumptions that have been used over the years in the modular HTGR business. Sources of data on radionuclide behavior that are used in that model, those models and the experimental methods for data collection.

For the next slide our July 6th letter requested NRC staff provide us some positions on functional containment and on mechanistic source terms. With regard to functional containment, we asked the staff to establish some options regarding functional containment performance standards for modular HTGR's. And this is analogous to what was requested by the Commission in the SRM, the SECY-03-0047 and is further discussed in SECY-05-0006.

These are topics that have been under discussion globally for quite some time in relation to advanced nuclear reactors. We also requested some positions on mechanistic source terms themselves. That the staff endorse, or at least find reasonable the definition that we use for mechanistic source terms, which is the quantities of radionuclides released from the reactor building to the environment during a spectrum of licensing basis events. That includes the timing, the physical and chemical forms of the release, the thermal energy to the extent that there is any and so on and so forth.

We'll get into that a little bit more in a little bit. Not yet, Mark. We asked for them to agree that the source terms can be calculated on an event-specific basis and determined mechanistically using models of radionuclide generation and transport accounting for the fuel and the reactor design characteristics and passive features. The actual physical performance of the various radionuclide release barriers and then to agree that we've identified the key HTGR fission product transport phenomena and established acceptable plans to evaluate those.

You'll hear more detail about the plans for evaluating those phenomena when Dave Petti gives his presentation on the AGR fuel program. We're not going to go into any great detail in this presentation on the details of how specific radionuclides move around through core materials. That's a little lower level of detail then I think we can get into today with the time we have available.

CHAIR BLEY: When you sent this letter, did you anticipate the answers to these requests to be in the evaluation of the White Papers or something separate?

MR. ALBERSTEIN: In the --

CHAIR BLEY: Should I ask staff about this?

MR. ALBERSTEIN: -- in the working group assessment report on mechanistic source terms that was done in February of 2012, the staff already found our definition to be reasonable and the general approach to be reasonable subject to how things go as we move on through the technology development program.

There were a number of items for follow-up that they identified in the assessment report. And many of those are items that will only be clarified and resolved as the AGR fuel program moves through completion.

CHAIR BLEY: Thank you.

MR. ALBERSTEIN: Okay. Moving on to some background, regulatory background. Issues of containment requirements and alternative approaches to calculating source terms that go beyond the traditional Light Water Reactor approach of a robustly tight containment and assumed release fractions consistent with TID 14844.

There's a long history going back to the late 80's and early 90's of regulatory staff and the industry taking a look at these kind of issues. It's addressed in the Advanced Reactor Policy Statement. It's addressed in a number of SECYs including some recent ones for advanced LWRs that talk about new, revised or physically based source terms.

And in the history of HTGR licensing in the United States for Peach Bottom, Fort St. Vrain, the large HTGR's in the 70's and then the DOE-sponsored MHTGR program in the 80's and 90's and the more recent Pebble Bed application, pre-application submittals both by Exelon and by the PBMR folks in South Africa, these topics have been under discussion for quite some time.

More specifically, for the next viewgraph, if one looks at NUREG 1338, which was the draft safety evaluation report written by the staff for the modular HTGR in the late 80's, it's noted in there that the staff judged that siting source terms can be based on mechanistic analysis, taking into account fuel failure and behavior of the radionuclides in the plant. It was noted that final acceptance of a mechanistically calculated source term was dependent on successful completion of R&D.

And we're in a similar position today. And that's pretty consistent, I think, with what I just described in the February 15th working group assessment reports. In 1995, the new reg 1338 was updated. And it was noted then again that a mechanistic source specific to the design is an acceptable approach.

The next slide gives a little more information from the 1995 draft of that NUREG noting that again that a mechanistic approach is a reasonable approach to use, subject again to the fuel performance being well understood, the transport phenomena being adequately modeled and the spectrum of events for which one calculates source terms being sufficiently founded.

DR. KRESS: Question.

MR. ALBERSTEIN: Yes.

DR. KRESS: Maybe this is pretty standard, but I'll ask you anyway. Does the position one takes on these various issues depend on the individual plant power level?

MR. ALBERSTEIN: I believe that the methodology that one would apply to do mechanistic source term calculations is independent of plant power level. One has to take that into account, obviously in doing the specific analyses. But the fundamental --

DR. KRESS: I was wondering if there was

going to be a limit based on the power level that you could have depending on one's position on these issues.

MR. ALBERSTEIN: I think from a designer perspective, the important objective is be able to meet the design goals of meeting the PAGs at the exclusionary boundary.

DR. KRESS: That would limit your power?

MR. ALBERSTEIN: There's certainly some power level at which it would be difficult to do that, yes.

MEMBER CORRADINI: Power level or power density?

DR. KRESS: Well I'm thinking power density, but --

MEMBER ARMIJO: There's always a lot of way to skin that cat.

MR. ALBERSTEIN: There's a lot of ways to skin that cat.

DR. KRESS: Yes.

MR. ALBERSTEIN: So there is certainly some limit or combine limit of parameters. None of the designs for modular HTGR's that have been looked at to date get any where near such --

DR. KRESS: Yes, I don't think there's a

problem with those.

MR. ALBERSTEIN: But, yes, certainly in theory there's a limit.

MEMBER RAY: Well relative to power level, I mean I've been trying to figure out where to ask a question about external event consumption. You know, we're all very aware of an external event that will effect multi-units, which is similar to the larger and larger power, it tends to aggregate multi-unit plants. What has been part of this analysis from that standpoint?

MR. ALBERSTEIN: I'm going to let Fred answer that.

MR. SILADY: I can give you the background on what we did on MHTGR. We took a look at this. An earthquake that we thought bounded at 85 percent of the US sites, 0.3 g, we assess that to be in the design basis event region, of about 0.3 g. Then we design for it. And this is for all the plants, all the reactors at the plant.

Then and in the beyond design basis event space, we looked at a more severe more, 0.7 g. And it gets very difficult to put a frequency number on it, site dependent and so on. But it was roughly in the 10^{-6} range. And we assessed it, best estimate on the capability of the plant to respond having been designed for 0.3 g and found that point on the chart and treated it accordingly in terms of top level regulatory criteria and so on.

But we didn't do anything with internal or external floods at that point. But the process would float in much the same way, same with fires and other things.

MEMBER RAY: Well the principle of modularity is to make each of the modules capable of withstanding an event itself. The question is when you're talking about offsite dose, what is the effect if all units at a site are effected by an event?

I understand what you just said, so you don't need to repeat it. But it doesn't add up to the same thing as simply assuming all the units have been effected by an event simultaneously. And okay. Can you answer the question, then?

MR. KINSEY: This is Jim Kinsey. Fred, you just want to maybe reiterate or go through the take away of the per plant year concept again to make sure that that's clear?

MR. SILADY: Well, I think the Committee understands that both on the frequency and consequent space the per plant year is key. And we try to pick up events that effect all the modules. But then they may each respond differently. Shut down cooling system on one may survive the earthquake and on others it may not.

So there's usually a split fraction when you come to the end that says, okay, we started off with an earthquake and at the last question he just asked looked at all the things that respond for that particular sequence. Did it effect one, two, three or four of the modules? And typically we find out that for an earthquake it effected all of them.

Some of them might have recovered with the shut down cooling system. And so the next likely thing is that it only happen in one. And the fact that it happened in two and three is usually pretty much percentages as opposed to 50/50 and it kind of depends on the event between one and four.

And so we're taking it into account in a thorough way. And if any of those events, one modular going through it, four going through, two or three, is in that design basis event region or close enough within the uncertainty bands we treat it as a design basis event.

DR. KRESS: Let me ask you a strange question. If you have multiple modules do you just have one control room and one set of controllers?

MR. SILADY: That was the design in the MHTGR, yes. And that --

MEMBER CORRADINI: Say that again.

MR. SILADY: Yes. One control.

DR. KRESS: There's just one control room and one set of operators.

MEMBER CORRADINI: No.

MEMBER SHACK: For the suite.

DR. KRESS: For the whole suite.

MR. SILADY: And that's part of our safety design basis of making it independent of operator actions. And we look at what he can do, what he has to do and what he could do that would be an act of commission and that's in the process. We've got work to do on the NGNP. But from what we understood from the MHTGR, for the events we looked at then and the conceptual design it met the requirements.

MEMBER REMPE: Two or four modules? How many modules?

MR. SILADY: We have four on the MHTGR.

MR. ALBERSTEIN: The specifics there would be subject to whatever the designer of the next plant comes up with. Okay. Moving on with regard to containment alternatives in the staff's evaluation of the modular HTGR in the 1995 update of the safety evaluation, it was noted that Commission had already noted in other proceedings that conventional LWR type containments are not an absolute requirement for advanced reactor designs.

That we could have containment functional design criteria to evaluate the acceptability of alternative approaches to containment instead of containment design criteria that are prescriptive. And that allows the acceptance of containments with leak rates that are not essentially leaktight per General Design Criterion 16 for light water reactors.

So the point of all of this is that alternative approaches to containment and alternative approaches to determination of source terms, not a new subject. It's something that's been kicking around for about 20 years. And what you're going to hear in the next segment of the presentation, if you would move to the next slide, is a little more detail about the modular HTGR approach to containment performance and source terms.

Next slide. We refer in our White Papers and in our presentations to something we call the functional containment. What is this thing? What it is, it's a collection of design selections that taken together ensure that the radionuclides are retained at their source in the fuel for our safety design strategy and that the regulatory requirements like 10 CFR 5034, 5279 for offsite dose and our plant design goals of beating the PAG's are met at the Exclusion Area Boundary.

The next viewgraph shows, is the same one that Fred used in his presentation showing you the multiple barriers in the HTGR that comprise the functional containment. It's a different approach to some extent from that off the light water reactor in that our emphasis on the importance of the barriers is from the inside out. Whereas for the light water reactor it tends to be a little bit more from the outside in.

MEMBER ARMIJO: David, I don't agree with that. I think it's, if you start from the inside out really the only thing that you have in your fuel particle that's the equivalent to cladding is the silicone carbide, very particle, hangs by itself, but the rest of this other stuff that would inhibit release, not prevent it, but would inhibit it.

Then the other barrier you have is the vessel and the steam generator tubes and no containment. And the reactor building is not a containment. And that's been an age-old issue whether you really need a containment if you truly have fuel with the characteristics of the TRISO fuel and its operated at a low enough temperature.

But I think you have all the elements if you get sort of distributed containment function. But you really only have one, I've seen your fuel on them and I only see one barrier and that's silicone carbide.

MR. ALBERSTEIN: I'll show you a viewgraph in a little bit that gets into the relative performance of those barriers for select radionuclides. But again our real focus here is to heat the radionuclides in the fuel thereby lessening our reliance on the downstream barriers.

MEMBER CORRADINI: You're going to get the reactor building eventually anyway, right? So we don't have to ask about what it is right now.

MR. ALBERSTEIN: We'll get there. So let's move on. Our definition, I already had it in an earlier viewgraph, we define the source term as the releases from the reactor building, the stuff that doses the public taking into account the timing which can for the most severe events consist of an initial release followed by a delayed release, physical and chemical forms and the thermal energy.

The real emphasis here is that we're focused on release from the reactor building. In a couple of viewgraphs I'll get into other aspects of source term that you're familiar with in the light water reactor world. Our approach is to calculate these on an event specific basis mechanistically using the models that take into account the behavior of HTGRs and the mechanical performance of the various release barriers.

That's different from the LWR approach of a source term based on a severe core damage event with perhaps fuel melting and so on and so forth. Next slide. When one does analyses to determine source terms in HTGRs, hundreds of radionuclides are considered. The exact number depends on the exact code.

I know in the case of the general atomic code that something like 250 radionuclides. PBMR might be a slightly different number, but a lot of radionuclides are in there and to facilitate their analysis it's been found to be useful to group them by chemical similarity or by similarity in their transport properties. That makes it a little bit more of a manageable problem.

Based on past analysis for HTGR accident sequences, modular HTGR sequences, we think that iodine-131, the two cesium isotopes, 137 and 134 and strontium-90 are going to be the dominant contributors to offsite dose. For the worker silver, silver-110 can be a major contributor. But for offsite it tends to be these four. So --

DR. KRESS: Ruthenium not in there?

MR. ALBERSTEIN: Not a big contributor. Not a big contributor.

DR. KRESS: You treat iodine as molecular iodine?

MR. ALBERSTEIN: Yes.

DR. KRESS: I guess it has to be.

MR. ALBERSTEIN: Yes. And we tend to conservatively assume that it has transport properties at least out until it gets into the coolant, similar to that of noble gases. So the models that are used do an analysis of the transport of the radionuclides from their point of origin through the fuel and into the circulating helium. They then do analyses to determine the amount of each radionuclide circulating within the helium pressure boundary.

Radionuclides like strontium, iodine, cesium, most of the metallics are condensable and so they can plate out on the surfaces. Within the helium pressure boundary that too is modeled. In the event of a breach of the helium pressure boundary they've modeled the release of radionuclides and distribution within the reactor building and then release from the reactor building to the environment.

So in the process of doing these calculations, you get information on radionuclide inventories throughout the facility. And those inventories can be used for other purposes, shielding analyses, worker dose, equipment EQ, control room dose. And those are all applications of source terms that are typical in the light water reactor community also.

We just don't typically refer to those as source terms in the same context that we refer to source terms as released from the reactor building. But you can do all the same things that the other guys do.

DR. KRESS: The condensable radionuclides, is iodine the only one?

MR. ALBERSTEIN: Chronium, cesium, silver, they're all --

DR. KRESS: They're not already condensed before they get out of the matrix?

MR. ALBERSTEIN: Most of them don't get out as you'll see in a viewgraph that's coming up. But to the extent that they do, they tend to plate out on surfaces within the primary circuit, within the helium pressure boundary. So the amount of those radionuclides that's circulating is --

DR. KRESS: I always thought once they hit the helium they would already be solid?

MALE PARTICIPANT: Yes, I think you're right, yes.

MR. PETTI: Condensable means that it can, not that it has necessarily.

MR. ALBERSTEIN: We're talking six, seven orders of magnitude reduction relative to what's in the reactor. Yes, okay. The next viewgraph is, I must give credit to this, this was developed by David Hanson at General Atomic who's sitting in the audience here. And if you guys have any questions I'm going to throw them at him.

But what this viewgraph is, is an attempt to capture in one picture all of the phenomena that take place with regard to fission product generation and transport within the plant systems and that need to be taken into account when doing modeling of source terms. Some of these phenomena occur only during accidents. Not all of them occur during all accidents.

You can see from the representation here it shows you a fission product being generated inside the fuel kernel. If the coatings are failed and in the case of a few fission products even if they're not failed, fission products can work their way out of the particle into the matrix.

You also have a certain amount of heavy metal contamination, residual heavy metal outside the particle coatings that results from the fuel fabrication process. So radionuclides can be generated by fission of that heavy metal. It represents the transport of the radionuclides through the graphite webbing in the case of a prismatic design. And ultimately the release to the helium pressure boundary.

Plate out the condensation of radionuclides on those HPB surfaces is represented down in the lower right. You see the effects of the helium purification system which is used primarily to ensure that residual oxidant levels in the helium are held to acceptably low levels. But that also has the advantage of removing some circulating radionuclides.

Under certain accident conditions like a moisture ingress you could get washoff or steam induced vaporization of the condensed radionuclides from the helium pressure boundary surfaces. That's shown down toward the bottom. In a rapid depressurization a certain amount of the condensed radionuclides could be lifted off of the surfaces as a result of the blowdown forces. That's shown also down in the lower right.

Then as a result of a leak or a breach, one could release radionuclides from the helium pressure boundary into the reactor building where they would be subject to condensation, deposition or settling as shown along the bottom. And then of course release from the building itself can happen either by virtue of intentional venting, in the case of a rapid depressurization or as a result of building leakage, which as was noted in Fred's presentation is typically assumed to be in the neighborhood of 100 percent, per day leak rate for reactor buildings. So this is just an attempt to capture all of these phenomena in one incredibly complex drawing.

MEMBER REMPE: How much uncertainty

exists now with the data for some of the phenomena that you've shown on this?

MR. ALBERSTEIN: It depends on the phenomenon you're talking about.

MEMBER REMPE: And how much data are out there, like do they run those tests at Comedie? There was a big flack with Rainer Moormarn, a couple years ago, right?

MR. PETTI: The dry Comedie experiments were completed. The wet ones were not, as I recall. So some of them were.

MEMBER REMPE: Is that an area where you feel like you need to have more data if things were ever to progress?

MR. ALBERSTEIN: Yes, and although we did not choose to go into it in this presentation, the White Paper has a discussion in it of fission product behavior knowledge gaps. And a discussion of the efforts that are planned within the AGR program to close some of those gaps.

MEMBER REMPE: And where would you do those tests? I'm sorry I didn't look at that section. But is that something you would be doing overseas or in the US or what's, hadn't thought through?

MR. ALBERSTEIN: Has not been identified where.

DR. KRESS: That atom identified as radionuclide dust interactions, is that graphite dust?

MR. ALBERSTEIN: Primarily.

MALE PARTICIPANT: This covers both technologies, so --

MR. ALBERSTEIN: Yes, this is supposed to be generic, both pebble bed and prismatic block, which is why --

DR. KRESS: You get more dust in the prismatic?

MR. ALBERSTEIN: No, you get more dust in the pebble bed.

DR. KRESS: From the pebbles.

MR. ALBERSTEIN: I mean in the AVR reactor there was a lot of dust in there. For prismatic designs to date, Peach Bottom, Fort St. Vrain and HTGR in Japan, there hadn't been any basically. So we think this is a phenomenon that is primarily of interest in pebble bed designs.

DR. KRESS: Are any of these phenomenon burnup level dependent?

MR. ALBERSTEIN: Well certainly the

performance of the fuel particle coating is dependent on burn-up. And the models that we use to determine particle performance and subsequent radionuclide release take that among other parameters into consideration.

MR. PETTI: The kernel, you would think the kernel obviously would be burnup-dependent.

DR. KRESS: So you do your source term calculation at the maximum burnup point? Is that the way you work that?

MR. PETTI: Equilibrium core, right?

MR. ALBERSTEIN: Yes, usually one looks at burnup histories for the entire core and add an equilibrium core configuration.

DR. KRESS: You never reach it for cesium and strontium.

MR. ALBERSTEIN: It takes into account that some parts of the core see more or less fluence and burnup and temperature then other parts and integrates the hole to get a total core release.

DR. KRESS: Do you have a calculation for each section of the core?

MR. ALBERSTEIN: Yes.

MR. PETTI: It's incredibly tedious.

MR. ALBERSTEIN: Let's move on to the

next slide. As noted in the earlier presentation, the coatings are the primary barrier to radionuclide release both during normal operation and off-normal events. I mentioned you have a certain amount of heavy metal contamination, heavy metal outside of particle coatings.

That and any initially defective fuel particles and as manufactured fuel can contribute immediately to radionuclide release outside of the core into the helium pressure boundary. And so typical specifications for this kind of fuel is that such defects and heavy metal contamination fractions are in the neighborhood of 10^{-5}.

I think in Dave's slides coming up you'll see some more specific numbers. One can't guarantee that not one of the billions of particles in the core will fail sometime during normal operation. So we have design goals for that also. Again, pretty small number. Something in the neighborhood of 10^{-4}. Likewise during various licensing basis events one can't guarantee that not one particle will fail.

So we have a design target for incremental release for an incremental fuel failure during licensing basis events, about another 10^{-4}. What we found is, in our analyses, is that radionuclide release during LBEs tends to be dominated by the exposed heavy metal. The contamination and the kernels that are exposed either during the initial fabrication process or the incremental exposure of kernels during normal operation or the transience.

DR. KRESS: That first bullet, that's one defective fuel particle in a hundred thousand?

MR. ALBERSTEIN: Yes.

DR. KRESS: And you're going to make your quality assurance of the manufacturing process such that you can meet that?

MR. PETTI: The specification is actually double that 2 10^{-5}.

MR. ALBERSTEIN: And Dave's going to show you a lot of information in his presentation on how we've been demonstrating that to date in the AGR field program and how that fuel's been performing relative to these kinds of expectations.

MEMBER ARMIJO: If heavy metal contamination resulting from fabrication is a significant contributor, what can you do about that in your fabrication process just to prevent it? It may not be necessary to --

MR. PETTI: No, I tell you, yes, I mean

the real reason why it dominates is because everything else is so low. So you're really down in the weeds.

MR. ALBERSTEIN: You're in the weeds.

MR. PETTI: And, you know, we're kind of at probably the limit of the technology at these levels. Plus there's other issues. I mean, let's say you wanted to do 10^{-6} on heavy metal contamination. To do that statistically, how much fuel you have to destroy to check, it's impractical and cost prohibitive.

MR. ALBERSTEIN: You wouldn't even be able to figure it.

MEMBER ARMIJO: So it's just your process control that says I guess all these coatings are put in, in the same coating reactor machine and so you start with uranium in there and some will still be out there when you're coating with the final layer. So there's no way you can prevent that.

MR. PETTI: But we, there's all sorts of things we do to minimize it. We change the inside of the furnace every coating and clean it.

MEMBER ARMIJO: You do that.

MR. PETTI: There's all sorts of things that are done, that's been learned over the years when they used to make 10^{-3} fuel. How do you get down to 10^{-5}?

MR. ALBERSTEIN: And I think we'll plan to cover a lot of that --

MR. PETTI: We'll cover a lot of that in the next session.

CHAIR BLEY: But just as an anchor for me I think in Fred's talk, you made the point that through monitoring throughout operation you're able to ensure that you haven't had any operational events that could have created situations that aren't what we're expecting should you ever have an accident?

MR. PETTI: You're monitoring the concentration of fission products in the coolant. And there are things called plateout probes to monitor the cesium for instance.

DR. KRESS: We'll have some sort of criteria then that says whoops that didn't make my specs. I better shut down and refuel.

MR. ALBERSTEIN: That's possible, yes. If one were to have some particles fail in an unexpected manner and to an unexpected degree, you'd see it in circulating activity and you'd see it fairly quickly.

MEMBER CORRADINI: Would you be able to

a like a light water reactor determine where?

MR. PETTI: No.

MR. ALBERSTEIN: Historically that has proven to be difficult for HTGRs.

MEMBER CORRADINI: So you'd have to completely refuel? I'm asking to make sure I understand the point. Because if you can detect it that's fine. But if you have to detect it and change out the whole core, goodness gracious.

MR. ALBERSTEIN: The economics of that are not in practice.

MR. PETTI: That's an owner concern.

MEMBER ARMIJO: But you, it's very sensitive, the temperature I would think, so.

MR. ALBERSTEIN: Somewhat, yes.

MR. PETTI: Somewhat, wait until you see some of the results. I think the phase base is much bigger then we think. I think there's a lot more margin then we think.

MEMBER ARMIJO: I would guess.

MR. PETTI: So we're finding some really, I mean, very exciting new things that suggest that there's a lot more room then --

MR. ALBERSTEIN: But in answer to your question, at least for a prismatic core, pinpointing an exact fuel element that has fuel that's giving you a problem is a problem. For pebbles, you pick it up in the refueling machine. You can pick it up in the process of circulating the pebbles. But for prismatics that's an issue.

Okay. Next viewgraph. So most radionuclides during normal operation will reach a steady state concentration because of their relatively short half-lives and a steady state distribution in the primary circuit. The long-lived isotopes like the cesium-137, the strontium-90, they are exceptions that plateout inventory builds up over plant life. And in anticipation of what you might ask, when one does the types of accident analyses and source term analyses that Fred was talking about, you assume in the plant life inventories of the long-life fission products like 137 and strontium-90.

And the concentration and distribution is effected by this list of parameters you see here, half-life, initial fuel quality, sorptivity on the various circuit surfaces. Those little ticks underneath that second bullet are basically in words the things that I showed you in the complicated picture. I think we can move on from there.

And I believe the next slide, yes, okay.

The next slide gives you, the next two slides give you examples of comparisons between calculated and measured fission product release in Fort St. Vrain for normal operation. This first slide shows you as a function of operating time measured circulating activity or R/B rather which is release-to-birth ratio, which is directly proportional to circulating activity for a krypton-85m throughout the plant life, that's the blue dots in the figure.

The solid line up above is what was calculated using the survey code at General Atomics as a function of time. And you can see that the calculated circulating activity was larger then that which was measured, it's a good thing. You can also see that it was within about a factor of four of the calculated, the calculated and measured were within about a factor of four of each other.

Historically at GA, it's a little bit different at other vendors, but historically at GA they have sought to be able to determine circulating activities within about a factor of four. And that objective was met for these analyses. You can see from the broken line that's the circulating activity that one would have predicted had there been no coated particle failure at all. If it was just due to release of radionuclides from heavy metal contamination.

Keep in mind that Fort St. Vrain's design was not intended to have the kind of minimal fuel failure the modular HTGR is intended to have. Dave mentioned a little bit ago that at Fort St. Vrain they were making what we call 10^{-3} fuel, the initial defects were a little higher. Maybe actually it was below 10^{-3} but not as good as what we're talking about for the modular HTGR.

MR. PETTI: Up to a hundred, that's not little.

MR. ALBERSTEIN: Yes, the point here is that these methodologies have been used to successfully calculate circulating activities. And in the next viewgraph --

MEMBER ARMIJO: Before you leave that just is there a way for let's say at the highest measured data point that you could extract from that the fraction of failed fuel particles?

MR. ALBERSTEIN: Yes, you could back calculate it.

MEMBER ARMIJO: Do you have any idea what that was? Is it like one in 10 thousand, 1 in five hundred or?

MR. HANSON: The best guess, the models we added in --

CHAIR BLEY: Please, come to the mike and state your name. Please for the record.

MR. HANSON: I'll learn to keep my mouth shut.

MEMBER ARMIJO: Don't worry about that.

CHAIR BLEY: Just come up and join us.

MR. HANSON: All right. I'm David Hanson. I now consult for Idaho. I served my time at GA for 40 years doing these things. At the end of life the, based upon these measured R/Bs the exposed kernel fraction was approximately 8 times 10^{-3}.

MEMBER ARMIJO: Okay. The design was five percent fuel failure for Fort St. Vrain and so you were at?

MR. HANSON: It's a different design basis. It's an example for the heavy metal contamination which is now 10^{-5} to these modern designs. For Fort St. Vrain it was 10^{-4}.

MEMBER ARMIJO: So it was less then one percent and your design goal was five --

MR. HANSON: Was five percent.

MEMBER ARMIJO: Okay. And then you're dropping that down a couple orders of magnitude for the MHTGR, is that?

MR. HANSON: Yes.

MR. ALBERSTEIN: Dave will show you how we do that when he gets to his presentation. The next viewgraph is a comparison of calculated and measured condensable radionuclide content at Fort St. Vrain. Again during normal operation for the strontium-90, cesium-134, cesium-137. These analysis were done also at GA using a code called TRAFIC.

The PBMR folks have other codes that do analyses of these types of radionuclides. And again you can see measured was less then calculated, a good thing. In the case of condensable radionuclides the metallic rate of nuclides typically at GA the objective was to get them right within a factor of ten. At PBMR it was a factor of five for some isotopes, a factor of ten for others and a factor of 20 for others. GA tends to shoot, tended to shoot just for a factor of ten and you can see the results are within that range.

The purpose of these two viewgraphs again is just to show you that there is the ability to do these types of analyses within acceptable degrees of accuracy in support of mechanistic source term calculations.

DR. KRESS: When we calculate source terms for light water reactors we didn't differentiate between cesium 134 and 137. Why is it different here?

MR. ALBERSTEIN: I'm sorry. I didn't catch that.

DR. KRESS: What is the difference between cesium-134 and 137 that makes them release different? We lumped them together in the light water reactors.

MR. PETTI: Well 134 is an activation product off of 133. So it's generation is a little bit, you know, different.

DR. KRESS: This is depending on the concentration that's in there.

MR. PETTI: Yes, this is an absolute curies again. This isn't a fraction. So there would be a difference.

MR. ALBERSTEIN: The first slide was fractional release. This one's straight curies.

DR. KRESS: I understand.

MR. ALBERSTEIN: Okay. Let's move on. Fred mentioned and I think I mentioned earlier, that for off-normal events one can have release of radionuclides in two phases. An early release and then a delayed release. Not all licensing basis events entail a delayed release, but for those that do we'll get into a little bit of the mechanisms.

For circulating activity the circulating around the helium pressure boundary within the helium pressure boundary, those can be released in a matter of minutes to days depending on the size of the break or breach of the helium pressure boundary. The amount that actually gets out depends on where the release takes place and any operator actions that might be taken for example to intentionally depressurize the system in the event of a slow leak.

For large breaks you get large shear forces within the helium pressure boundary as the helium is depressurized out through the breach. And in those situations where the shear force on a given surface, on a given location within the helium pressure boundary becomes higher then the shear force during normal operation, some of the condensed radionuclides can be re-entrained and subsequently released from the helium pressure boundary. Again, the amount depends on the size of the break and therefore on the size of the shear forces within the helium pressure boundary and on the location.

For certain accident scenarios like a

moisture ingress, a sufficiently large moisture ingress can result in the lifting of the pressure release valve which is another contributor to early release. And moisture ingress can result in washoff of a certain amount of the radionuclide content that is condensed on the helium pressure boundary surfaces.

The relief valve may cycle open and close or it may fail open depending on the exact scenario you're looking at, all of those are mechanisms that contribute to the early release for certain off-normal events and need to be taken into account. And in the case of a rapid depressurization event which raises the pressure in the reactor building, those radionuclides that initial burst of pressure is intentionally vented from the building to the environment.

This being an acceptable strategy hinges upon being able to manufacture the fuel with low levels of contamination, low levels of initially defective fuel particles and operate the reactor with low levels of incremental fuel failures such that when one releases that vented release to the environment, the objective of meeting the PAGs, design objective of meeting the PAGs at the EAB and the regulatory requirement to meet 5034 of the EAB is still met.

That hinges back to what Fred presented earlier showing the importance of fuel fabrication quality relative to the safety design approach.

MEMBER CORRADINI: So maybe if we, should we wait asking about if the geometrical configuration that allows that because you'll come to it later?

MR. ALBERSTEIN: The geometrical configuration of?

MEMBER CORRADINI: What I want to ask is, is this building vented and filtered or is this building just vented?

MR. ALBERSTEIN: We'll come to that in a minute.

MEMBER CORRADINI: Good.

MR. ALBERSTEIN: Actually we, I'll go for that now. We received an RAI, I believe, on this subject during the staff's review of the White Paper. And the building designs to date have been simply vented. And whether one would go beyond that in future designs is an issue that the designer of the next plant is going to have to address.

MEMBER CORRADINI: So is that a nice way of saying you don't want to put it in a box?

MR. ALBERSTEIN: Yes.

MEMBER CORRADINI: But and if this is the wrong time to ask it, is it not unreasonable to say given where we are historically that's not a defendable position?

MR. ALBERSTEIN: You have to look at the specifics of the plant and the source term behavior of the specific design.

MEMBER CORRADINI: But I, okay, well.

MR. ALBERSTEIN: There have been some alternatives on reactor building design configurations for the PBMR and I believe also for the prismatic designs. And this is one option that's been looked at --

MEMBER CORRADINI: If you're going to come back to it, I'll wait.

MR. ALBERSTEIN: I'm not going to get into any quantitative stuff.

MEMBER RAY: Let me interrupt you and say because this bears on something I've been trying to follow and I'm not sure I can. In the first presentation the phrase defense in depth approach was used twice. And then in the second presentation it was said that the defense in depth philosophy is by maintaining multiple barriers against radiation release.

MR. ALBERSTEIN: That's part of it.

MEMBER RAY: And by reducing the potential for consequences of severe accidents. Now what, at least the way I take Mike's question --

MEMBER CORRADINI: Take it any way you want at this point.

MEMBER RAY: -- is the containment building a part, a defense in depth barrier or is it not? That's a simple enough question that there ought to be an answer to it.

MR. ALBERSTEIN: The reactor building does provide some attenuation for radionuclide release. We're going to jump ahead to the punch line in one of my later slides. Analyses that have been done so far indicate that relative to meeting the regulatory requirements at the exclusionary boundary, 5034, 5279 --

MEMBER RAY: I think we understand that, I mean.

MR. ALBERSTEIN: -- yes, we don't need it.

MEMBER RAY: It's well presented, understand it. The question is what do you mean by defense in depth?

MR. PETTI: There's defense in depth with barriers and there's programmatic defense in depth. Speaking with the barriers, the building provides some retention. Does it provide enough is the question?

MEMBER RAY: No, that's not the question. I'm asking a simple question. Not efficacy, but what do you mean by defense in depth?

MR. KINSEY: This is Jim Kinsey. We could spend a couple of minutes on this. But we transmitted a White Paper to the staff that gave a pretty extensive discussion of our defense in depth proposal. That wasn't a part of the series of staff positions that we've asked for feedback on. So we don't have an extensive presentation on that topic today.

And I'm not sure if we'd be able to fit an extensive discussion into the time that we have. But I think it's an important topic and we'd be happy to, you know, talk about it maybe in an alternate session.

MEMBER RAY: All right. That's fair enough. I just --

MR. KINSEY: I'm not trying to turn off the discussion. I'm just not sure if we're --

MEMBER RAY: No, that's a satisfactory answer. I just, I can't discern from what you're talking about, which I do understand, what you mean when you refer to defense in depth. The best, closest thing I can come to it is this phrase here which I don't understand. So let's put it off and --

MEMBER CORRADINI: Let me ask another question then you can say we're not going to talk about it or it's going to come later. Is the reactor building part of an SSC? Is it, in light water reactors the containment is part of a system safety component that we need. Is this thing that?

MR. ALBERSTEIN: Let me answer that. You're asking is the reactor building safety related.

MEMBER CORRADINI: Yes.

MR. ALBERSTEIN: Okay. Yes and no. The yes is because it is necessary and sufficient to provide the structural protection of the helium pressure boundary and reactor, reactor cavity cooling system. We rely on it. It's made safety related for that function.

The no part is traditionally it's been safety related for light water reactors to retain radionuclides. We don't need it to do that. We do need it to meet the PAGs. But we don't need it to meet 10 CFR 5034. So we want to put the focus on the reactor building on what we, is necessary and sufficient.

MEMBER CORRADINI: You need it from a source term reduction standpoint for your PAG levels?

MR. ALBERSTEIN: That's correct. Yes.

MEMBER CORRADINI: The efficacy or the quantitative value can wait on, but okay. I'll just stop there for now.

MEMBER ARMIJO: Do you really need it to meet your?

MR. SILADY: The MHTGR, they asked for the mean, the upper bound, all the different possibilities with and without. And we concluded that in one or more of the beyond design basis events, it was required and it was the water ingress one for the PAGs which is our design goal.

Is it required for the NRC safety goals, no. Is it required for the design basis events for 10CFR 5034, no. Is it required, you know, so we classified it safety related. But we really want to keep the focus and the effort on protecting of external events and maintaining the core geometry.

MS. BANERJEE: Are we taking these as an action item for the April 9th presentation then to go more over the defense in depth?

MR. KINSEY: We would be happy to provide a further discussion on defense in depth in that session. But I guess we'd need to talk about that and decide if we can fit that discussion in the day and still leave the staff time to present their outputs on these other topics. So maybe we can talk about that in the wrap up today. We're certainly willing to support it, we just need to I guess manage everybody's time.

MS. BANERJEE: Is there a desire to hear more?

CHAIR BLEY: There will be, the staff will have, in the staff's responses to the White Papers there's a White Paper on defense in depth. And that will have been reviewed and we'll see that.

MR. CARLSON: We did review that and that appears in our working group assessment from February of last year. And the updated assessment report will be updated, but there won't be extensive updating under that topic.

MEMBER RAY: I don't have any problem saying we don't need defense in depth or defense in depth means something different then what you think it means and here's what it means. I just want to understand what we're talking about here.

CHAIR BLEY: I guess in response to Harold's question some presentation on what you mean by defense in depth, there is a White Paper on it. It's appropriate and I have to admit I'm a little confused. There's two different questions. What is defense in depth and if there were containment would that be defense in depth? And to me that seems obvious it would be. But I don't think that's what they mean by defense in depth, so.

MS. BANERJEE: Level of defense in depth and how it's met.

MR. KINSEY: So we can take an action then to do a short presentation on the topic in the April meeting.

CHAIR BLEY: I think so and we'll hear from staff also on their evaluation of the White Paper. So go ahead, please.

MR. ALBERSTEIN: Okay. Next slide. Delayed release mechanisms that have to be modeled. Obviously those things that contribute to release during normal operation, contamination, defective particles, particles that fail in service would continue to contribute during an off-normal event.

Historically we've found, not yet, we've found that for those accident scenarios which have a delayed release and they tend to be the dose-dominant scenarios, the delayed release is typically larger then what you get from release of circulating activity and any amount of liftoff or washoff. It depends on how much time the fuel spends at temperature.

Coated particle fuel, as you saw from one of the slides in Fred's presentation, doesn't just go to pot. The coatings don't just fail at some temperature threshold. It's a time at temperature phenomenon. It needs to be taken into account. It's also affected by the level of oxidants in the system and by the volatility of the specific radionuclide you're talking about.

And again, the delayed release is a function of location and size of breach of the primary system. And then the timing relative to the heat up and cool down of the core. And as Fred mentioned, a small leak actually has a greater release, can have a greater release then a larger release, then a larger breach from the helium pressure boundary.

And I think we already touched on the rest of these sub ticks, except the last one that once the temperatures within the helium pressure boundary decrease as the core cools down, then the releases will cease eventually as the core gets to lower temperatures and there's no further driving force to support the release.

The next slide is a representative presentation of functional containment performance during a depressurized loss of forced cooling event. And what we're trying to show you here is the relative effectiveness of each of the barriers of the functional containment in the retention or attenuation of fission products throughout the functional containment.

So taking as an example the green bars which show iodine production and release, the particular analysis here was for the modular HTGR in 1989. You got 10 million curies roughly of iodine-131 in the core to begin with. That which gets out of the fuel is attenuated by about four quarters of magnitude.

In the models it's assumed that whatever gets out of the particles also gets out of the graphite and into the circulating activity. So you see no attenuation in the next step. And then that which can get out of the, out of the graphite into the primary boundary under this particular accident scenario, it's attenuated by another factor of about 20, by the helium pressure boundary itself.

And then you can see in the last step there's another attenuation factor of about a factor of 10, which is what's provided by the reactor building.

MEMBER ARMIJO: And that's for a reactor building that can exchange all the air in --

MR. ALBERSTEIN: That's for the reference design.

MEMBER ARMIJO: The reference design. So it's a leaky building.

MR. ALBERSTEIN: Compared to an LWR, yes.

MEMBER ARMIJO: Okay.

MR. ALBERSTEIN: On the other hand, for cesium and strontium you can see again the retention by the fuel particles is about the same. But the amount that's attenuated by the matrix and the graphite material differs, the retention factors differ when you compare cesium to strontium.

But again overall we're talking about retention of radionuclides by six to eight orders of magnitude which is consistent with the safety design approach of retaining radionuclides at their source.

So a lot to chew on here. But that's a brief --

CHAIR BLEY: But just a couple simpleminded questions from me if you would. The charts, is this the result you're showing here consistent with the experiments you've run in the first two boxes I guess, the first two columns that Dave will be talking about later?

MR. PETTI: No, these are higher.

CHAIR BLEY: These are higher. The basis for these were?

MR. PETTI: Back in 1989, what they thought --

CHAIR BLEY: Okay that's what this is that we're looking at.

MR. PETTI: So we would show you that the release from fuel is even better.

CHAIR BLEY: And would there be any difference if for the next step if you had pebble bed or if you had prismatic? Do we know?

MR. ALBERSTEIN: I don't think I could speak to that.

MR. PETTI: Which step?

CHAIR BLEY: Released from the graphite.

MR. PETTI: Yes. That's complicated. There's a little bit and there's some canceling factors. I think it's in the same order of magnitude.

MEMBER ARMIJO: Will you be showing us a chart, the same kind of chart with modern fuel, fuel you've been testing?

MR. PETTI: We should have been. No, what I will show you though is relative to the requirements. So you saw some discussion on incremental failure. I'll show you what that means. So it won't look at the curies. But there will be some relative ratios that will translate directly.

CHAIR BLEY: And this is a depressurized loss so when you get to the last two boxes getting out of the reactor building there's no driving force it's just air circulation?

MR. PETTI: Well there's the initial release that's in --

MR. SILADY: But that's not dominant here. And you're right there's, in fact it depends on the timing. It may be sucking nuclides back in to the helium pressure boundary.

CHAIR BLEY: Go ahead. I was just trying to figure out what we were really looking at here.

MR. ALBERSTEIN: There's a lot of data here. And we were trying to find a relatively succinct way to present it. Move on to the next slide. In summary on functional containment our emphasis again is on radionuclide retention within the fuel during normal operation with the release of a relatively low inventory of radionuclides to the helium pressure boundary.

The limiting off-normal events tend to be characterized by an initial release from the helium pressure boundary that's a function of leak size, break size, pressure relief performance and so on and so forth. And then a larger delayed release from the fuel.

Our analyses thus far have indicated that this functional containment, this overall system of barriers, will meet the regulatory requirements for offsite dose at the EAB with margin for a wide spectrum of off-normal events without even taking into account the retention factors of the reactor building. But to meet the EPA PAGs at the EAB with margin, we do need to take into account the retention of radionuclides by the reactor building. So moving on.

MEMBER CORRADINI: So, I'm sorry that I have to get pulled out. But just let me repeat what you said before and make sure I didn't mishear. So with your last, your fourth bullet the reactor building is filtered or is not filtered in your current design concept?

MR. ALBERSTEIN: It is not.

MEMBER CORRADINI: And Fort St. Vrain was not also?

MR. ALBERSTEIN: It was not.

MEMBER CORRADINI: Okay. I guess my memory banks say it was but, okay.

(Crosstalk)

MEMBER REMPE: It had a what?

MR. PETTI: An HVAC, heating, ventilating and air conditioning but it wasn't available in DVA number one.

MEMBER CORRADINI: Okay. So it was there but it wasn't called upon in the analysis?

MR. PETTI: We will have heating, ventilating and air conditioning in the reactor building too.

MR. ALBERSTEIN: But it's not --

MR. PETTI: But if you go to the frequencies of these things, it often times isn't there.

MR. ALBERSTEIN: And I guess on this point of whether the reactor building is filtered or not I think what we're trying to present here is the process and the definition for a mechanistic source term.

MEMBER CORRADINI: I'm with you.

MR. ALBERSTEIN: I think we recognize that the reactor building is one of the attenuators of a release. And it will be in the details of the final design as to whether the --

MEMBER CORRADINI: I understand but just so I remember it though, the attenuation is occurring by physical processes without a filter?

MR. ALBERSTEIN: Yes. There have been various different designs over the decades, some have had filters.

MEMBER CORRADINI: I just wanted to verify that.

MR. PETTI: That's largely the longer the delayed release is what's getting attenuated. That initial venting, there's nothing that, you know, it's not in there long enough, so.

CHAIR BLEY: But you haven't been heated up.

MR. PETTI: Right.

MEMBER ARMIJO: Just before you leave this, when you say at the last bullet you can meet these requirements for a wide spectrum of off-normal events does that mean all the identified off-normal events that you're designing for? What is the one that's not, you're not capable of providing this meeting of PAGs?

MR. ALBERSTEIN: We've not yet found it. But we will in the next presentation, talk about bounding event sequences which go beyond design basis.

MR. SILADY: To clarify though it's the design basis events and the beyond design basis events meeting the PAGs.

MEMBER SHACK: But it's with your current definitions of what a beyond design basis event is?

MR. SILADY: Correct.

MEMBER ARMIJO: Got it.

MR. ALBERSTEIN: Okay. I think I have only one more slide, well don't count that one. I have one more slide. And this is sort of our overall conclusions. Number one, we believe that the approach to functional containment and mechanistic source term being taken for modular HTGRs is consistent with the Advanced Reactor Policy Statement.

It's consistent with discussions of

containment function and mechanistic source terms in a wide variety of SECY documents that have been issued over the last 20 years. And it's consistent with approaches that have been previously reviewed by the NRC staff for modular HTGRs, particularly for the MHTGR reviews in the 80's and 90's. It's also consistent with the approaches in the pebble bed reviews that were done roughly ten years ago.

We take an event specific approach that can be applied to the full range of licensing basis events using mechanistic models for fission product generation and transport accounting for the inherent behavior of HTGRs, their passive design features and the mechanical performance of the fission product release barriers that comprise the functional containment. And that's all I have on that topic.

MEMBER RAY: You got one more slide according to this.

MR. ALBERSTEIN: Do I?

MR. PETTI: I did too and I missed it. I set the pattern here.

MR. ALBERSTEIN: Okay. We'll get this right the next presentation. This is just a summary recap of the things we've requested the NRC staff to give us positions on. Shorthand summary of the ones that were in the earlier viewgraph. So now I'm done.

CHAIR BLEY: And now you have another.

MR. ALBERSTEIN: Yes, the fun doesn't stop. As I said at the beginning of this presentation, the next presentation, this presentation on siting source terms in somewhat a subset or a specialized aspect of the overall topic of miscellaneous source terms.

So if we can move to the next slide. That's where we are in the agenda. And next slide. I'm going to talk a little bit about the staff position regarding site and source terms that we've requested. We're going to talk about the approach to be taken to siting source terms.

Then we're going to talk about a further specialized aspect of this which is event sequences involving graphite oxidation. We're going to get into that because it's been the subject of a lot of discussion between the project and the staff over the last several months. So we felt we should address it here. And then we'll give you some conclusions overall on siting source terms.

CHAIR BLEY: For the issues you're going to talk about here, they're not covered in your mechanistic source term White Paper, are they or are they embedded in there?

MR. ALBERSTEIN: They're really somewhat separate.

CHAIR BLEY: You don't have a White Paper on this one?

MR. ALBERSTEIN: No, we don't. These are issues that have come up since the staff issued its working group assessment report on February 15th. And as I've said, we've spent a lot of time on this with the staff. And it's garnered quite a bit of attention and that's why we wanted to give you a presentation on it here today.

So requested position, next viewgraph. A lot of words here. But siting source terms, in the light water reactor community, are developed based on an assumption that one looks at an accident sequence that entails a substantial meltdown of the core with subsequent release of pretty large quantities of fission products. That comes up in the footnotes to 10 CFR 5034, 5279 and one of the subsections of 10 CFR 100 in the earlier days of the regulations.

And taking that language that talks about melting of the core and applying it to a reactor that number one has no metal in the core, number two has taken a safety design approach to ensure that relying only on passive behavior of the plant, the fuel isn't going to reach temperatures at which significant fuel particle coating failure can occur. That creates for us a bit of a dilemma.

How does one address this regulatory requirement in the context of an HTGR? So what we requested is that the staff develop a position to give us some final determination of regarding how licensing basis events would be considered for the purpose of plant siting and functional containment design decisions. Taking into account that the staff has previously found that improved fuel performance is a justification to revise siting source terms and containment design requirements.

Next slide. So what approach are we going to take here? Well the approach that we plan to take is patterned after that which was developed back in the late 80's and the early 90's in the modular HTGR review. This approach was documented both in the PSID and the PRA for that reactor. And the findings regarding their approach were discussed in the staff safety evaluation NUREG-1338.

So the first step is to develop the design consistent with the safety design approach that Fred has already described and to utilize risk insights as input to design, to the design for the range of requirements, both user requirements and regulatory. And then select and mechanistically evaluated LBEs including the DBEs, DBAs and BDBEs against the top level regulatory requirement and against our design goal of beating the PAGs at the EAB.

And I'm consistent with the mechanistic source term approach do a mechanistic evaluation of these events that have limiting dose consequences, the highest dose consequences offsite and use those source terms as the siting source terms. Go to the next slide, give you a little more information.

Fred already showed you that for the MHTGR they identified three design basis accidents that were the highest offsite dose consequence with the limiting DBAs. And Fred's already gone through the brief description of each of these. It's interesting to note that each of these entails ingress of either moisture or air into the reactor. So if one were doing this for that old design, these would be the limiting design basis accidents that would be used to generate siting source terms.

MEMBER RAY: This is a single tube rupture in six, is it?

MR. ALBERSTEIN: Correct. That was a single tube rupture. And we're going to talk about a scenario with more tubes in just a second. We can go on to the next slide.

During the review of the modular HTGR by the staff back in the 80's and 90's, they started asking a series of questions about, I think the attempt here was to try to determine what kind of margin one has in this plant given that approach to selecting limiting design basis accidents. And the staff postulated a number of bounding event sequences, and we'll show them to you in just a second here, to try to test just how far the plant could be pushed while still meeting the regulatory requirements and the design goals in the way that GA and DOE were attempting to do at that time.

So we would take elements of that from that review and use them in siting source term determination today. Specifically what we would do is that to ensure that there aren't any cliff edge effects out there where things could go bad unexpectedly and to understand just how much margin we have, we would supplement the LBE-derived siting source terms with insights from a best estimate mechanistic evaluation of some bounding event sequences.

Don Carlson has referred to this as a survey of the safety terrain, just to see what's out there. And I think that's a good way to describe it. But this isn't a free for all. This isn't an opportunity for people to exercise their imagination and come up with exotic scenarios that don't make any sense.

Number one, they need to be physically plausible rather than just non-physical, arbitrary combinations of event parameters. Now physically plausible is a subjective term and it's intentionally subjective here. It means you don't pretend that the laws of physics have suddenly been suspended in order to come up with some exotic accident scenario.

It means that you don't suddenly assume that the physical properties of the materials in the core are radically different from what they're known to be just to try to create some kind of large release. They have to be physically plausible. They have to be sensible. And we'll give you some examples in a minute here.

These are event sequences that are in 10 to the minus double digit frequency range. And it's pretty hard to rigorously quantify frequencies when you get down to those kinds of numbers. But nonetheless we do expect that generally these kinds of sequences would have frequencies lower than the BDBE region which has it as it's minimum cut off 5 times 10^{-7} for plant year and frequency.

As we evaluate the events we're going to consider again the intrinsic and passive behaviors of the HTGR. Next slide. So how do you determine what the bounding event sequences would be? You do a deterministic and this is a purely deterministic process, by the way. You do a deterministic review of plausible events that potentially impact the safety functions of removing core heat, controlling heat generation and controlling graphite, controlling chemical attack, for example graphite oxidation.

In order to do an initial selection of bounding event sequences you have to have your design fairly well established. You have to be through preliminary design. But we can say at this time that the bounding event selection process we'd use as a starting point. The six bounding event sequences that were requested by the NRC staff back in its review of the MHTGR.

And if we go to the next viewgraph you'll see what those were. And I'm just, you can read these for yourself. I'm going to highlight a couple of them okay.

MEMBER CORRADINI: These are the ones that the staff back in '89 asked you guys to consider?

MR. ALBERSTEIN: Yes. That's what they are. One of them was in inadvertent withdrawal of all control rods without scram for a 36 hour period. And the 36 hours is the number that the staff came up with at that time. I think it was a reflection of their thinking at that time that within 36 hours some kind of action would be taken to mitigate the consequences of a sequence like this.

I know that in today's world we're talking about different lengths of time, longer lengths of time. That's okay. But at the time this was what they were working with. So inadvertent withdrawal of all control rods is one example.

Number four, steam generator tube rupture that takes out 25 percent of the tubes with failure to isolate or dump. Number five, a rapid depressurization of one module resulting from a double-ended guillotine break of what they call the crossduct, it's actually a Section 3 cross vessel with a failure to scram and an assumption that the reactor cavity cooling system is failed or unavailable for 36 hours. And thereafter only 25 percent of it is unblocked.

So these are pretty bounding, severe types of events, okay. They all resulted in low offsite doses. They all, when GA did its exit accident analyses for these sequences they still resulted in the ability to meet the PAGs at the exclusionary boundary.

So this gives you some sense, that gives you some sense of, that there's quite a bit of margin here. In terms of frequency assessment, all most all of these were in the ten to the minus double digits regime, which one would expect also.

So we would use these as a starting point. But whatever we eventually choose as the bounding event sequences, we can go to the next slide, but what would we do with the results of the analyses? The applications would be number one, they'd be used to identify and understand the potential for cliff edge effects, for high consequence events.

Theoretically, we may find something. We may find something that has more risk then we thought we did, that we had in this. There might be something out there. And so it's an opportunity to determine potential risk significant plant or system vulnerabilities. And if we do find a cliff out there to identify risk mitigation strategies, most likely design changes, that we would have to implement to address that.

And the result of all of this is going to be documented as part of the license application process. It's not clear where in the structure it would go, whether it would be topical reports, Chapter 19, it's really not Chapter 15 material. But it would get documented as part of the record.

There have been some previous staff positions taken on bounding event sequences. In the 1989 version of the draft safety evaluation, the staff indicated that it judged that these bounding event sequences they had proposed, the results of those analyses showed that the MHTGR had the potential to cope with these rare, severe events without the release of a significant amount of fission products.

The ACRS also in the safety evaluation back in those days, noted that neither the designers nor the staff or the ACRS members themselves had been able to postulate any accident scenarios of reasonable credibility for which additional physical barriers to the release of fission products would be required to provide adequate protection of the public. The additional barriers might include a filter on a vent. At least at that time, there was no need identified.

Move on to the next slide. I'm going to talk a little bit about graphite oxidation event sequences. And the reason we want to talk about this is that back in 1993 the Commission issued a staff requirements memo stating that the Commission believed that for the MHTGR the staff should be addressing an event entailing the loss of primary coolant pressure boundary integrity whereby ingress could occur from the so-called chimney effect and we'll talk about what that means, resulting in a graphite fire and the subsequent loss of integrity of fuel particle coatings.

You'd have to oxidize a lot of graphite for that to happen. But the staff at that time believed or somebody on the Commission at that time believed that was a scenario that should be looked at. And we've had quite a bit of discussion with the staff over the last ten months, 11 months or so about what should be done to address this old SRM.

There are previous staff positions that have been taken with regard to oxidation event sequences in the modular HTGR. In the draft NUREG-1338, it reflected the results of some independent analyses that Brookhaven had done in support of the safety evaluation. They noted that for graphite oxidation to proceed to a point that structural damage inside the core could be possible, you'd have to have an unlimited supply of air for many days.

You'll recall from comments Fred made earlier, for these depressurization events that do result in some air ingress, it's not pure air that gets in there. It's a mixture of air and helium and it is in fact mostly helium. So Brookhaven concluded that you'd have to have an unlimited air supply for many, many days. And in the 1995 update to the safety evaluation, the staff concluded that a graphite fire in the MHTGR is a very low probability event.

They also noted that as stated in another NUREG done by one of their contractors, without two breaches of the reactor vessel to create a chimney effect, one up high and one down low, it's not likely that significant amounts of air will enter the core and therefore that graphite fires are not a licensability issue for the modular HTGR. To get a chimney that's going to really move the air through the core, you've got to have a breach of an ASME Class 3 vessel at the top and you've got to have one down low.

And I think all of you guys who are light water reactor people will recognize that's a scenario that goes far beyond the types of scenarios that have typically been required in reactor safety regulation. Onto the next slide.

MEMBER ARMIJO: You don't have any kind of penetrations at the bottom of that vessel?

MR. ALBERSTEIN: At the bottom.

MEMBER CORRADINI: No, they have them in the crossvessel.

MEMBER ARMIJO: Just the crossvessel. I mean but really down at the bottom.

MR. ALBERSTEIN: At the bottom it's a shut down cooling circulator down there.

MEMBER ARMIJO: Okay. That's the only --

MR. ALBERSTEIN: Yes. Go on to the next slide. So what's the approach that we would take today to event sequences involving graphite oxidation? First of all consistent with the findings of the staff in NUREG-1338 and with the findings of the ACRS back in the 90's, we really think that the frequency of the kind of event described in the SRM is going to fall so far below the LBE spectrum that the event would be considered incredible. We think this is a ten to the minus double digit type of event.

Those expectations have to be confirmed for the specific design of the next modular HTGR. And that will be done. However, we recognize that bounding event sequences that maximize the potential for graphite oxidation do need to be considered, even if it's not that particular scenario from the SRM. And it's the intention to consider those in the bounding event sequence process as part of the NGNP design and licensing effort.

There are data needs in the area of both air and moisture effects on core materials. When Dave gives his presentation here he'll talk a little bit about our plans in the AGR fuel development and qualification program to address the effects and obtain additional data on the effects of air and moisture ingress.

Next slide. Conclusions, next slide. Number one, the approach we're going to take is essentially the same for siting source terms, essentially the same as that proposed in the days of the modular HTGR review in the late 80's and early 90's. I believe this approach is consistent with discussions of containment function and mechanistic source terms and more recent SECY documents and what approach is previously reviewed by the staff. Limiting LBEs will be evaluated to determine SSTs and physically plausible bounding event sequences, including some involving graphite oxidation will be considered to make sure there are no cliff edges.

DR. KRESS: What exactly does that mean?

MR. ALBERSTEIN: Pardon.

DR. KRESS: I'm not sure exactly what that means because if you're going to include a graphite fire.

MR. ALBERSTEIN: We will look at sequences that entail graphite oxidation. You see I'm judiciously avoiding the use of the f-word.

DR. KRESS: I don't mind. But you're going to look at it and, you know, depending on how much air you get in there it could have devastating effects. So are you going to look at it, from what standpoint? Limiting the amount of air or?

MR. ALBERSTEIN: We'll look at what we believe are bounding ultralow frequency events that entail air ingress to assess their effects on the performance of the system under such accident conditions.

DR. KRESS: I'm still not sure what you're going to do.

MR. SILADY: We'll probably do what we did with the MHTGR.

DR. KRESS: I assume it's such a low frequency.

MR. SILADY: No, we looked at, maybe we can go to the backup slide, Mark, on graphite oxidation if it's easy. Otherwise I'll just do it verbally. He's going to pull it up. And I think if you see it as well as I say it, there's a better chance of communication.

It's number twelve. That's it. So some of these things we've already talked about. Graphite will oxidize with the oxygen in the air or in a helium/air gas mixture. The nuclear grade graphite is much less reactive then other types of graphite due to its graphitized structure and high purity. The oxidation of the graphite is limited by the amount of air in the helium gas mixture from the reactor building.

And then once that mixture comes in the

high flow resistor in the coolant channels to the core height, talking prismatic here, has an L/D greater then 700. So it's hard to get air to go up. It's going to react with these core support posts if it's at the bottom or wherever it comes in. Fuel particles are embedded in the graphite matrix. We've talked about that.

And those little compacts are within the fuel element. See you have to oxidize away a lot of graphite to even get to the fuel particles which of course have a silicon carbide layer on it. Loss of all forced cooling and depressurization of the helium pressure boundary are required for air to get, to ingress to begin with in the mixture.

Sometimes we forget that. You've got all these days and you have to not turn on any forced cooling and cool the core down. And you have to have a leak or a break of some size in the helium pressure boundary. The chimney effect was mentioned in the SRM. I suspect that if you have a really large opening you get stratified flow as well.

But the point is, it is a very large opening and you've lost forced cooling. Maybe those are synonymous if it's that large. We did some analyses. This is what we would probably do. This answers your question, what are you going to do? Well when we looked at a 22 square foot opening at the bottom of the vessel system, we didn't do a double- ended guillotine break. But we did the size of a vessel at the bottom of the vessel system. And we only got one percent of the core graphite oxidized after 30 days.

And that amount of air is equivalent to eight reactor building volumes of a hundred percent air. That's what we did in the 80's. And the reason it was only one percent of the core graphite oxidized is because we didn't get any additional decay heats or heat generation from the exothermic reaction. But the oxidation did not lead to a loss of core geometry. And it was limited such that we didn't incremental radionuclide release due to oxidation.

This by the way was two orders of magnitude at one percent. We got 10^{-4} fraction from design basis events. That was going factor of a hundred greater. More recently NGNP analyses have shown that a break in the helium pressure boundary leads to a very small percentage of air in the gas mixture after the helium blowdown.

So not only did it take 22 square feet, not only would it take eight reactor building volume, so not only did we look at it for 30 days, but we were assuming a hundred percent air.

Those analyses on a different reactor building should have been more like two percent air and 98 percent helium. We'll have to do this same sort of thing, but we'll be fighting this for a long time I'm sure. But we have to change the perceptions of whoever's writing the SRMs.

MEMBER ARMIJO: Fred, where did the 22 square feet come from? What did you have to do to create that kind of a --

MR. SILADY: We had to fail the vessel either more plausibly it would probably be around the weld of the cross vessel to the reactor vessel or the cross vessel to the steam generator vessel. That was it, that stratified flow.

MEMBER ARMIJO: Well did you ever fail the penetration at the bottom of the post?

MR. SILADY: No, we didn't.

MEMBER ARMIJO: I think that would be more likely then the cross vessels.

(Cross talk)

MEMBER RAY: Are the core support posts too low in temperature to be concerned about?

MR. SILADY: No, they can oxidize.

That's where most of the oxidation takes place. If you get two moles for every mole. And so it sort of chokes itself at some point trying to get up through that cooling hose because the hot wants to rise.

MEMBER RAY: But it doesn't threaten the function of --

MR. SILADY: We didn't find that we lost any posts and the core is latticed such that one post doesn't cause the core geometry anyway.

MEMBER RAY: I've got to go to another meeting and so do you.

CHAIR BLEY: Lots of us have to go to another meeting in just a couple of minutes.

MEMBER REMPE: Quick question. Earlier you talked about the seismic analysis. And in many cases you can rely on the various data. But are there some specific components that are HTGR specific and are there data for those HTGR specific components?

MR. SILADY: With the response to a seismic event?

MEMBER REMPE: Right. To give insights on how you've quantified your seismic analysis.

MR. SILADY: I think a lot of the structures are unique. I mean they don't have graphite core support posts.

MEMBER REMPE: And they never did any sort of seismic, so is that another data need that's been identified in your documentation?

MR. SILADY: Well this event, it's in the 0.3g sited source, safe shut down earthquake. It's designed for that with margins. And then we looked at it for a 0.7g which is more a 10^{-6} level. And looked to make sure it had the capability with its embedded below grade damped configuration.

MEMBER REMPE: What about a prismatic fuel element? Anyone ever tested that to see what happens with vibrations is kind of what I'm kind of getting to? I mean are there a lot of things, are there any data to help justify. You can design for it, but --

MR. ALBERSTEIN: As I recall, it's a long time ago, but as I recall there was a seismic response testing program done for Fort St. Vrain where they did some shaking of simulated smaller versions of an HTGR prismatic core.

MR. SILADY: The Japanese did some of this too.

MEMBER REMPE: That's what I was wondering. I didn't know how much specific data there was.

MR. ALBERSTEIN: There's some information out there.

MR. SILADY: I think it was in the PBMR plans as well.

CHAIR BLEY: Okay. Do you have any more you want to do or is that it?

MR. ALBERSTEIN: Was there one, the only slide that was left was a recap.

CHAIR BLEY: Okay. I don't think we need that. We're going to have to break for some of us to go to another meeting. We'll all be back here at one waiting to hear from Dave. That sounds pretty interesting. So we're recessed until 1 o'clock.

(Whereupon, the foregoing matter went off the record at 12:13 p.m. and went back on the record at 1:01 p.m.)

CHAIR BLEY: The meeting is back in session. And I'll turn it over to Dave Petti. Is that right?

MR. PETTI: Yes. We'll talk about --

CHAIR BLEY: It's been a while since you've actually been here, quite a while.

MR. PETTI: Yes, yes, yes. So --

CHAIR BLEY: Welcome back.

MR. PETTI: Yes, we'll talk about where we are on fuel qualification and fission product behavior with a snapshot of the program. Next slide. And yes, we are the last item on the agenda.

So I'll talk about what the White Paper, the requested NRC staff positions are. We'll go to background real quickly, talk about our approach to qualification. And I pose a number of simple, key questions that will help focus us as we go through the presentation and then talk about the program plan status and the key results, particularly as they relate to licensing.

So this is not sort of my typical fuel talk. This is sort of inside out instead of outside in maybe. So it's to look at it from the licensing sort of perspective. And I'll go through each of the pieces of the program and talk about what it means for licensing, what we've learned, where I think we're going to end up because we're still in service and then sort of a summary and what's it look like in the future.

So the White Paper shown there is submitted in July 2010. And the staff position is to confirm that the plans being implemented by the program are generally acceptable, provide reasonable assurance of the capability of the coated particle fuel to retain fission products control in a predictable manner. And particularly to identify any additional information or testing needed to provide adequate assurance of this capability.

So what we're really looking for is there some big multi-million dollar, multi-year thing we're missing? You know, little things you can incorporate along the way. But if there's something big, we really want to know now because this stuff, this work you know, takes a long time and a lot of money.

MEMBER ARMIJO: Dave, before we go on I'm going to have to leave early and there's one burning question I want to leave with you and you can answer whenever it's appropriate. I've gone through your White Paper and a lot of the stuff you've done, fine work that the laboratory has done. But fundamentally you fabricate this fuel with batch processes, maybe they're large batches I don't know.

But it's a batch process. And you're talking about hundreds of thousands, maybe millions of fuel particles to make up a core. And so it would be multiple batches. And the question I want to ask is

what is the NDT or inspection technique that assures you that each batch is at the same quality standards when, you know, if a batch had 10 thousand failed particles in it I would suspect that wouldn't meet your criteria. But if the batch is 500 thousand particles, how do you know? So somewhere along the line just tell us how you ensure that you --

MR. PETTI: When are you leaving?

MEMBER ARMIJO: I have to leave at two.

MR. PETTI: I think we'll get there.

MEMBER CORRADINI: If we limit our questions.

MEMBER ARMIJO: That's my one question.

MR. PETTI: Next slide. I think you all know what the fuel looks like. Comes in, what I call two flavors, either compacts or pebbles. We've been focused heavily on compacts largely because when we started this program over a decade ago there was very healthy programs in pebbles internationally in China, South Africa and Europe.

Also just a factor for consideration, testing compacts is physically easier, they're smaller items. Pebbles are big. It can be done but it really limits what you can do. So next slide.

I think if you haven't go this message by now from the previous presentations it's the fuel, it's the fuel, it's the fuel. We have to demonstrate that we can retain the fission products and the key principles are that we can make high quality low defect fuel and characterize it in a repeatable, consistent manner.

And so we'll talk about that. And that the performance with very low in-service failure rates is achievable within the envelope, the operating envelope and the accident envelope and that we can calculate that performance to the requisite level of accuracy.

We are using a UCO which is a shorthand. UCO, uranium oxycarbide is a mixture of uranium dioxide and uranium carbide, both UC and UC2 are acceptable. This enables better performance at higher burnup than UO2. And particularly it suppresses a failure mechanism in UO2 known as kernel migration where the kernel moves and can potentially threaten the coatings. Because of the thermal gradients this is more important in prismatics because there are bigger gradients there.

You don't get any carbon monoxide formation. Chemically you gather the free oxygen produced from fission by the carbide phase so that when you fission you free up that oxygen and it reacts with uranium carbide to form more UO2 instead of reacting with the carbon buffer to form carbon monoxide which is what happens to UO2.

And so the internal gas pressures are reduced relative to the UO2 TRISO that the Germans and the rest of the world is looking at. And the fission products are largely immobilized as oxides. This is an engineered fuel form to tie up the fission products largely as oxides. And you can get longer more economical fuel cycles.

And just as a hint, I don't think I have it is because of the work that we've done the rest of the world is starting to look at UCO.

DR. KRESS: Quick question. UCO is a mixture of these. What percentage of each in this mixture?

MR. PETTI: Twenty-five percent carbide, I believe, 75 percent oxide. But we have a pretty, you don't, you have a good range. You don't have to hit it on the dot. You just need --

DR. KRESS: It doesn't have to be that precise.

MR. PETTI: It doesn't have to be that precise, no. But so the exciting thing for us is that the work that's being done here, people always thought UCO was very far off. And in fact it is now being looked at by both the Koreans and the Chinese, good performance because they see what it can do in terms of the economics of the system.

DR. KRESS: Chinese have an operating reactor?

MR. PETTI: They have a little ten megawatt and they're building, they just put concrete in their 250 megawatt pebble bed.

MEMBER CORRADINI: Where is that going to be located? Is it at INET just north of Beijing?

MR. PETTI: No, it is a separate --

MEMBER CORRADINI: It's on the coast somewhere?

MR. PETTI: It's on the coast somewhere.

MR. ALBERSTEIN: Twin unit.

MR. PETTI: Yes, it's a twin unit. So our approach to qualification establish a spec. We have specifications on the kernels, on the coatings and on the compacts. We implement a process capable of meeting that spec and implement statistical quality control procedures to demonstrate the spec is met. Unlike our LWR fuel we are not measuring on every particle, obviously.

DR. KRESS: What are your specs on the

coatings?

MR. PETTI: I'll, we'll talk about that. Then we test under irradiation a statistically significant quantity of fuel and with the monitoring to know the in-pile performance and the PIE to demonstrate that the requirements that we'll talk about are actually met.

Do the same thing under accident conditions and then use this data from the program to either improve the models or to qualify the models. So we have separate experiments to improve and then the qualification data come from a completely independent data set.

So these are the simple questions that we'll answer. What are the reactors designer's quality and performance requirements because then I'll show you what I think, how we're doing relative to that. Can the fabrication process meet those requirements? And will the fuel be able to meet the performance requirements under normal and accident conditions?

How well do the models predict what's being observed? You saw a little bit in Dave's talk. I'll have a little bit about what the new fuel, what we think is going on. And I'll also try to tell you a little bit about what we've learned. And we'll put those answers to those questions in red in the presentation so they jump out at you. Next.

So this is, you know, we don't actually have a design that we're going to present at this point. This is a based on historical MHTGR designs peak fuel temperature of 1400 C, a time average maximum of 1250. The canonical 1600 C under accidents that everyone knows about this fuel. Burnup of 18 percent, fast fluence lest then five, 10^{25}.

Now here's the quality specifications that come from the reactors designer. These are the major defect specifications. The contamination we have missing the defective buffers, missing a defective pyro carbon, defective silicon carbide, missing or defective pyro carbon. And then the in service failure rates under normal operation in core heat-up accidents.

And I highlight the ones that are really important that we can talk about today. The contamination and the defective silicon carbide are large drivers of the source term. Contamination is a, you know, a uncontained uranium. So the fission products from that would release.

Defective silicon carbide, the cesium will get through defective silicon carbide. Pyro carbons will not be a good enough barrier. So you tend to worry about those. And then those incremental failures.

DR. KRESS: What kind of defects can you have with silicon carbide? Is it the thickness of the layer or the density of this?

MR. PETTI: It can be density, porosity is probably --

DR. KRESS: Porosity.

MR. PETTI: -- and I will show you today that is not what we worry about in the field today. We're meeting, we're exceeding that specification by a factor of three to five at 95 percent confidence. I'm not at all worried about bad silicon carbide. But I think this has to do and we'll talk about how we make it. Technology today versus what the Germans did is really good.

And then I want to talk about the, we'll talk about the incremental failure rates 2 times 10^{-4} and 6 times 10^{-4} and where we are. We think there's a lot of margin there relative to the reactor. So here is where we spent a lot of time, I showed the process in a very simple overview. The top part is the laboratory scale process.

The bottom is the engineering scale. It starts with making kernels in a Sol-Gel process. I couldn't put all the pictures at B&W. They're our vendor where they actually make the kernels. It's a very involved process.

MEMBER CORRADINI: So B&W does both lab and engineering?

MR. PETTI: No, Oak Ridge did the laboratory --

MEMBER CORRADINI: That's what I thought.

MR. PETTI: -- and B&W did the industrial scale. A lot of work on getting the UCO and making really good UCO. And we've got that process. Then you go to coating. And the laboratory scale, I call it Coke can coating. It's a 60 gram charge, about the size of your Coke can, is the active cylinder.

You wouldn't, you inject gases that decompose, acetylene, propylene for the carbon layers. You form a carbon on the particles. It's a fluidized bed. For silicon carbide you use methyltrichlorosilane, hydrogen and sometimes argon. And in fact we're using argon as our base coating.

The industrial scale is bigger. It's about a two kilogram charge in a six inch coater. By comparison the Germans used an eight inch coater, five kilogram charge. That's what the Chinese still use. There are sort of trade offs in processing about what's the right size, you know. It depends on how big the capacity is for the plant.

MEMBER ARMIJO: Just to get it straight a coater puts on the silicon carbide?

MR. PETTI: The carbon layers and the silicon carbide sequentially, each layer.

MEMBER ARMIJO: And what's the size of the batch, is that the two kilogram?

MR. PETTI: Two kilograms, yes. That's the two kilograms uranium, not even --

MEMBER ARMIJO: How much fuel does that make?

MR. PETTI: In terms of number of particles?

MEMBER ARMIJO: No, no, yes, why not.

MR. PETTI: Millions.

MR. SILADY: But relative to your question, burning question, there is like eight or nine billion in a reactor. So this makes a million, this two kilograms.

MR. PETTI: So those, I'm not sure we're going to get, yes, go back for a minute to answer his question. So these are the specs that you worry about, do you have a good fuel. These are at the batch level and there are specifications at a lot level where you amalgamate batches.

MEMBER CORRADINI: And you sample to determine --

MR. PETTI: Yes, so here to meet 2 times 10^{-5} heavy contamination at 95 percent confidence I have to take a large amount of fuel.

MEMBER ARMIJO: And what do you do? What do you do, dissolve it?

MR. PETTI: Okay, then you go through a process called leach, burn, leach. You leach it in acid and that gets the easily exposed uranium. Then you burn all the carbon off, then you leach it again. And so if there's a defect in the silicon carbide the acid will go through and leach out. So you get the contamination and the defective silicon carbide.

MEMBER ARMIJO: Okay. So that's destructive and characterizes a batch.

MR. PETTI: Yes.

MEMBER ARMIJO: What about the actual silicon carbide integrity? Is that also with that leach, burn, leach?

MR. PETTI: Yes, if you had bad silicon

carbide the acid would get in and you'd know it. You also, we have specifications on all the thicknesses all the densities.

MEMBER CORRADINI: But that requires an inspection of a different sort then --

MR. PETTI: Yes, you basically take some particles and you section them and you have a computer that will calculate the thicknesses. That's what we do there. And then we also have anisotropy specifications on the carbon layers. I think those are the, all of the major.

MEMBER ARMIJO: Okay. So there's no nondestructive?

MR. PETTI: We continue to look and try to develop those. But the more you look, you know, you go back to this because you know it works. It's really hard. We have not been able to develop something that is, unfortunately. A lot of the effort in the last two years for us has been making compacts.

MEMBER CORRADINI: So can I ask another site question, Dave? So have you asked the staff to comment on your fuel sampling to meet those specs or is that yet to be done by whomever chooses to be the owner, operator of this thing and order the fuel?

MR. PETTI: I believe our White Paper talked about that approach and we had several RAIs on it.

MEMBER CORRADINI: But staff as part of their current activities was not to review and comment specifically on this.

MR. PETTI: No, I think they did.

MR. KINSEY: I guess as a point of clarification as Dave said, we covered that topic in our White Paper. The staff asked us some questions through some RAIs that we responded to. You know and if there was a concern about the process we're using for sampling we would expect that they would tell us that as part of our overarching question over are we missing anything.

MEMBER CORRADINI: Okay, thank you.

MEMBER ARMIJO: Just to make sure each batch is tested to meet the spec, that's what you said?

MR. PETTI: Yes. So go back, to make the compacts is really challenging. You only need 400 thousand pebbles, 450 thousand pebbles in a pebble bed of 600 megawatts. You need like three, six million one inch compacts. So the throughput is much different. So we started with the German process for overcoating and pressing. And they use like a Betty Crocker mixer to overcoat.

And it just was a multi-step process that when you looked at it from our throughput just wasn't going to work. So we had to develop a really completely different approach. And in fact it's just been very, very successful. You have to make this matrix and that's a complicated, it's a graphite flour and you put in the resin and you mix it together.

And we decided to go with a dry jet-milled product that the fuel vendor could buy from a supplier so that he didn't have to have large amounts. It's a carbon dust basically which you don't really don't want to necessarily have to do yourself. And very uniform which is really important, I think, in the overcoating.

Then we went and we bought a overcoater that the pharmaceutical industry uses to overcoat pills and the contact. That picture there, it's a large armoire size, is their lab scale, which just talks about our medical industry in the United States. They have much bigger ones. That's a production unit for what we're going to need.

MEMBER ARMIJO: Is there a binder or

anything or is that just dry?

MR. PETTI: Yes, it's a binder. Water, we that's another interesting story. The Germans use methanol. We didn't want to use methanol because it's flammable in a fuel facility. We tried water and it worked. I mean it stuck, it held it together. And you'll find that the overcoater, so you put the fuel in, put the matrix in, you put the water in 100 percent yield and the particles are better then what you get.

MEMBER ARMIJO: So you squish them and dry them and away you go?

MR. PETTI: Yes. It's great. Then you press them. We have an automatic presser. We can do about four compacts in 90 seconds. So you just, you'd triplicate this, automatic feeding. And then there's some heat treatment. So really nice. So we now have a full pilot line.

MEMBER ARMIJO: And this is at B&W?

MR. PETTI: This is at B&W. So, next. So we have basically reestablished the capability to make this fuel since last time the MHTGR was around. A lot of effort in understanding how to fabricate it which we think is really important. That it really isn't an art, that it is a reproducible. There is some science and engineering.

And today we're fabricating high quality, low defect fuel. We can meet the physical specifications and we're almost meeting the design of the defect specifications. The heavy metal contamination is at two 10^{-5}, sometimes we're at 2.5, sometimes we're at 3, sometimes we're at 1.8. We're in that mode trying to ring that out, particularly at 95 percent confidence.

We think with larger sample sizes and as we continue to mature the process we should be able to meet the defect specifications in a true production mode. The Chinese have done it. The Japanese have done it. We certainly don't think it's a problem. We have a vastly improved quality reproducibility and process control and characterization of the fuel.

One of the things that was important was control of the process. We use mass flow controllers with the gases and that gives you really nice, tight control that didn't exist when the Germans did it. We removed every high-variability human interaction in the process. We do not table these particles like the Germans did. We thought that we, we did a lot of work and found we were throwing away good material.

And we went with precision sieving which is used in many, you know, industrial settings.

So we've tried to bring today's technology to bear. And you see it. I mean the other one is making the silicon carbide. We're using an evaporator from the chip industry. You know, they've spent billions probably to make chips. You see it. We just see a better, you look at the cross section, the micrographs, it just looks better. You can tell it's really good. And it just has to do with where technology is today.

So we think establishing this vendor and the associated understanding really lends some credibility that what the Germans did in the 80's is repeatable and has a sound basis. And so all the technologies for this pilot line are in industrial hands and we'll be making the final qualification fuel in 2013.

So let's turn now to performance. This is, I've shown this before our radar plot for the five key parameters. Just a note that the brown curve is what we're trying to do. The dark green is the Germans and the light blue is the Japanese. So we do have a more aggressive performance envelope then historically done. But we have in fact been testing at these conditions and then having had success.

The packing fraction is higher then the last time I showed you because we've moved, based on design recommendations moved to a higher packing fraction.

MEMBER CORRADINI: And the packing fraction you define as the ratio --

MR. PETTI: Particles to the binder, to the compact volume, yes. So this is the program in one slide. Eight experiment campaigns AGR-1 through AGR-8. They each have a different purpose. AGR-1 was the laboratory scale fuel. AGR-2 is what we call performance demonstration. You could call it a dress rehearsal prior to the official qualification. It is a large coater, industrial scale fuel.

AGR-3&4 is got fuel that will fail in reactor to deal with addressing source term issues. We'll talk about that. Five, six is the qualification tests. Seven and eight are for validating fuel performance codes and fuel margin testing and then fission product behavior validation.

And then beyond the irradiation there's a parallel campaign with that irradiated material to do safety testing and PIE. And then moisture and air ingress effects are part of AGR-5&6. We plan to develop furnaces that we can put air and moisture quantities in and test the performance of the fuel, the fuel in the graphite body irradiate a graphite body even to do those sorts of tests.

Without doing anything in pile with large amounts of moisture. It's been done in the past and it's just not a big deal. Moisture is not really a problem with the fuel. So in terms of AGR-1, a slightly different particle.

We started with 350 micron UCO TRISO, 19 percent enriched. Goal burnups were 18 to 19 percent. We exceeded that a little bit. We went to about 19.5, 19.7. Peak time average temperature less than 1250. Average, volume average temperature may be around 1150. It took almost three years to do the irradiation and we had a very healthy population of particles, 300 thousand reached burnup of 19 percent with no failures.

They were tested in six individual capsules shown there and they each were individually controlled on temperature and that control gas is swept out into fission product monitors so if a particle fails you see a gas release, I mean we know if that's the case. So it took the Germans 15 years to accumulate the statistics to get that many because they were testing one, two, maybe three pebbles at a time. The volume of testing that we have in the ATR allows us to test a lot of particles very quickly, which is really good.

Okay, the next is a plot of the temperatures, the temperature census in the each of the six capsules shown on the right. And then temperature distribution expected, this is the conceptual design that GA did for the NGNP called the SC-MHR. And if you just look at it where it's, the experiment is a much more conservative in terms of the average temperatures of the capsules were somewhere between a 1000 and 1100. The average temperature in the core sits around 725, it looks like.

And then the peak temperatures, we've got a very large amount of fuel out at peak temperature much greater than you could expect in an NGNP. Next. This is another slide that not everyone has seen. This is relatively new so Don, you should be looking at this. We've taken the temperature predictions for, we basically have a finite element model and every finite element is about a particle. It tells you how detailed it is.

And we've basically created a time at temperature census. So this plots how much of the fuel was at what temperature for how much time. So take a look at the purple at 1300 degrees you can see that at a hundred hours about ten percent of the fuel saw temperatures in excess of 1300 degrees for a hundred days.

DR. KRESS: Each one of those dots represent a kernel?

MR. PETTI: Yes, basically each represent a particle in the test.

MEMBER CORRADINI: And the line represents what?

MR. PETTI: The line is sort of the average of those colors, the software will put like a --

MEMBER CORRADINI: So it's the average of the population at a time.

MR. PETTI: Yes, it's the average of that color, that strip. So you can see in the lower corner the little red. We saw five percent of the fuel, maybe three percent of the fuel greater than 1400 for 50 days. So this fuel saw a lot of time at high temperature. And so --

CHAIR BLEY: And irradiation at the same

time?

MR. PETTI: Yes under irradiation. And that's why I think some of the, we're seeing a lot of silver release. This is really because of the time and temperature.

So in terms of AGR-2, again this is vendor produced both UO2 and UCO at the time. We decided to, we had such success with AGR-1 that we would look at some pebble type fuel, 500 micron UO2. This South African fuel produced by South Africa, fuel produced by the CEA in France, part of our Gen 4 collaboration. We made UO2 at B&W. They can make either. Not a problem. It's one, you just don't put the carbon in and it's pretty much the same process, change centering schedule.

DR. KRESS: Going back to your previous slide, you don't have to go to, why does the temperature decrease, is it because you're using up the uranium?

MR. PETTI: We're holding it constant. But why would there, yes, this has to do with the detailed operation.

DR. KRESS: How you operate the system?

MR. PETTI: Yes, at the very end of the experiment when the temperatures are dropping because there is no uranium, we're trying to keep them up by changing the gas mix. And we have some controls in ATR to change the power, move some reflector and we kind of overshot. We kept saying we've got to keep the temperature up and we ended up running it actually very hot at the end so.

So notice the UO2 425 micron UCO, two capsules at 1250, one at 1400. We're calling it an early margin test. This is really an AGR-7 objective but we had the space. With the recommendation made from GA at the time and we thought it was a good one. And then the UO2 is much more pebble bed, 9.6 percent enriched, 11 percent FIMA. The French enrichment is, that what's they had so that's what we tested.

So it's a mixture. But it's really nice in this capsule each one's a different conditions. But we can do this.

DR. KRESS: Is the pebble bed then going to have this migration of the kernel problem?

MR. PETTI: The gradients in the pebble bed probably not as much, yes.

MEMBER ARMIJO: David, all these different fuels that you've tested, did you test them in the form of compacts or as particles?

MR. PETTI: Compacts.

MEMBER ARMIJO: Compacts. So they sent you compacts or --

MR. PETTI: The French sent us compacts. The South Africans sent us particles and we compacted them. So it's a particle test more than a --

MEMBER CORRADINI: Pardon my, that I don't know the unit or I forgot it. Remind what FIMA is.

MR. PETTI: Fissions per initial metal atom. Think atom percent.

MEMBER CORRADINI: And the enrichment is weight you're saying?

MR. PETTI: Yes, yes.

MEMBER CORRADINI: But what I want to understand is when the number is at or below the enrichment, I'm okay. When it's higher does that mean I'm doing some transmutation and burning?

MR. PETTI: Well, yes, you're doing that anyways even if they're lower ones, but, yes, yes.

MEMBER CORRADINI: Okay. But one is atom percent and weight percent to process it.

MR. PETTI: So this is a plot of the gas release the R/B. You heard that earlier. Think of it as a release fraction. I thought all the old US experiments post Fort St. Vrain there and all the

German experiments on the right. The blue crosses are the six capsules for AGR-1. So clearly as good as the German fuel at twice the burnup, German fuel would go to about nine to ten percent FIMA.

And the little box is where we are today on AGR-2. It's higher because there is an exposed kernel, a defect in every, again at industrial scale in almost every capsule. And then the hot capsule, the gas release from a hotter particle you get a little bit more gas release.

MEMBER CORRADINI: Did you tell us what you did to make a defect or that's coming? I forgot. I know you've told us in the past. I just forgot.

MR. PETTI: We believe that upon unloading of our coater we're damaging particles. And that's what the defect is. It's not inherent in the process.

MEMBER CORRADINI: Well I guess I'm asking about the pink. Are you answering the blue?

MR. PETTI: No, I'm answering the pink.

MEMBER CORRADINI: And so you consciously did that?

MR. PETTI: No, no. That was inadvertent. Inadvertently, how you take them out of the, this is a, it's an issue, it's in the handling between the coating and the compacting. And we're in an HEU facility. And I just can't tell the guys to go change out this thing because they live under very strict rules so everything's treated as HEU even though it isn't. It would take me a year to take a valve out of the system because it's part of their, a permanent part of their system.

MEMBER CORRADINI: So you because they're treating it as if it's HEU you think they're damaging it?

MR. PETTI: Well just the physical configuration. Some of the stuff that's there, so I said we'll get rid of that. And they said no, we can't. I mean it just, so in a real process line you could design from scratch you won't have that issue, you know.

MEMBER CORRADINI: It's avoidable.

MR. PETTI: It's avoidable, clearly an avoidable. So we're trying our best to work around it. That's basically, so --

MEMBER ARMIJO: But it just shows how sensitive it is to --

MR. PETTI: Yes, yes. Once you get them compacted they're great. But you've got to get them compacted. Now let me turn to the source term. I think you've heard this. We're going to have a mechanistic source term taking into account all the different values. The goal of our program is to provide the technical basis to support the design and licensing. And there are three experiments for two major campaigns called AGR-3/4 and AGR-8 and the follow on PIE safety test and loop testing. And AGR-8 is the independent validation part of the plan.

DR. KRESS: Will we see the details of these fission product models, eventually?

MR. PETTI: Not, maybe eventually, not today. But --

DR. KRESS: I mean I knew not today.

MEMBER CORRADINI: They gave us a hint at it in past presentations. You were there.

MR. PETTI: And I think in the White Paper there's some discussion in the appendixes maybe. I'm trying to remember.

MR. ALBERSTEIN: There's an appendix with a fair amount of detail on transport mechanisms.

MR. PETTI: A little bit on the RAIs.

MR. ALBERSTEIN: And later in Dave's presentation he's going to talk about a couple of these aspects.

MR. KINSEY: Excuse me, Dave, before we

move on. This morning we talked a little bit about the inside-out look versus the outside-in look. I was going to maybe ask either Fred or Dave to spend a minute on that and make sure that what we were communicating was clear as we're going through this if that's all right?

MR. SILADY: Yes, I'll be happy to just add some footnotes here maybe. I'm not a LWR guy by any means. I'm just a CGR. So I don't know comparisons. And oftentimes it's best to stay away from comparisons. But you know our barriers and we know that the fuel is the most important and it's receiving the most emphasis. And the silicon carbide is the most important barrier.

And we work inside out in that sense. We put more, tighter requirements on that fuel as opposed to helium pressure boundary or certainly to the reactor building. My understanding on the existing reactor barriers they have a clad, they have a pressure boundary and they have a containment.

And certainly if they have a problem with the clad they still have the helium pressure boundary there and they still have the containment there. But if they have a problem with the helium, not the helium pressure boundary, but the reactor coolant boundary their clad is linked to that in the sense that you have to keep the core covered or the clad goes. And then you've got much greater release that the outer barrier, because it's passed the coolant boundary's barrier, the outer barrier has to do yeomen's work to meet the requirements.

In our case, if we have a problem with the helium pressure boundary, it's okay. We still have the radionuclides in the fuel. There's no linkage. We don't have to keep the core covered with helium. It will operate at pressurized or depressurized, circulated and if it has any pressure it will be natural convection otherwise we'll heat up and we'll cool down. And the linkage isn't there.

And we only see a small fraction, a small increase coming from either the initial or the delayed release that goes into the reactor building. And so it's been designed to do the functions it needs to do, which is more focused on structurally protecting from external vents rather then being a radionuclide barrier.

So this concern about well you only have one thing left and it's not as good because it's a vented building and it's not a containment. I don't think that analogy is the right way to think about it because we don't have that ability or our silicon carbide to fail when we lose helium out the helium pressure boundary. So I just wanted --

MEMBER ARMIJO: In case you got the impression that's what I was saying, that's not what I saying. You just basically have a system that doesn't need three independent barriers because of the nature of your overall system. I don't have a problem with that. Just, but there aren't three, there's only two. There's the silicon carbide and your pressure vessel which are the only, what I call physical barriers.

MR. SILADY: I think we have more independence in our barriers then --

MEMBER ARMIJO: I think you have great fuels so don't, so let's not argue about it.

MR. SILADY: Good. Thanks. I feel better.

MR. PETTI: So let me turn back now. So this is the first source term experiment in the program AGR-3/4 to understand the behavior of the fission products from that small fraction of defected fuel. How much retention is the graphitic components in the core? And we use something called designed-to-fail fuel. So this is fuel, so we have a known source of fission products. So these are particles with no silicon carbide and very bad pyro carbon that we know will fail under irradiation.

DR. KRESS: That's how you designed it?

MR. PETTI: That's how you design the kernel.

MEMBER CORRADINI: You cooked them to be that way.

MR. PETTI: Right. We made them --

MEMBER ARMIJO: You fabricated them without silicon carbide.

MR. PETTI: Right and very anisotropic pyro carbon. So it rips itself apart under irradiation.

MEMBER ARMIJO: Okay. So this is about as bad as you can get?

MR. PETTI: Right. And then we, very carefully Oak Ridge developed a technique to put them all in the center of the compact. It was really cool. So we know that right, so we know that temperature. So we know, you know, really well. And then they even X-ray radiographed them so we know they're all on the center. And they all failed over Christmas. They always, it always happens over Christmas. I get the phone call, you know.

So and the goal here is to establish the transport of the metallic fission products and the retention in the graphitic components and the release from exposed kernels as a function of burnup, temperature and fluents. So we have, there you see the capsule. The inner ring is the fuel. The tannish striped one is an annulus cylinder of matrix material.

Then the outer one, the silvery grey is the fuel element graphite, the block graphite. And then the outer graphite with the holes is a sink that is there to, so no fission products go beyond and also for our instrumentation to go in those through tubes.

MEMBER CORRADINI: So the thought is the junk leaks out of the center hole to the outer ring?

MR. PETTI: Right. One dimensional diffusion as best as possible.

MEMBER CORRADINI: And it is captured there?

MR. PETTI: Right. Now we're monitoring for fission gas. So we're getting a release as a function of time which can, you can call to burnup and fluents. And there are 12 of these capsules stacked. So AGR-1 had six. This has 12 in the, what's called the northeast flux trap in ATR, a different location. And so we're able to get a really broad range of temperatures and burnups. And the matrix is quite complicated to show. But it kind of, we're trying to envelope the core that we'll get different combinations and be able to establish the functionality.

MEMBER CORRADINI: And these are shorter irradiations because you know as you cook them to temperature and fluents they're going to fail early on?

MR. PETTI: These failed in the first week as expected. Yes.

MEMBER CORRADINI: And then you're going to hold them there --

MR. PETTI: And then you hold them at that temperature.

MEMBER CORRADINI: For not again, not for three years though?

MR. PETTI: No, for about, it's about 400 to 450 full power days, which may be 18 months to two years in the reactor.

MEMBER ARMIJO: Dave, did you ever deliberately do any experiments in which you took the silicon carbide, didn't make it to full thickness just, you know, stopped coating it at a certain point and then test that fuel to see how long it would perform just to get a feel for the margin that you have in the retention?

MR. PETTI: We've never made, but for instance the Japanese silicon carbide is only 25 microns instead of 35 microns, showing very, very good behavior.

MEMBER ARMIJO: Yes, I would guess, the reason I'm getting to that point is that in a batch process when you do the qualification you, at least in light water reactor fuel, you go off-normal, you test your process controls and say well the lowest temperature I could ever be at would be somewhere in here. The highest I'll ever be within my process control range. Did you, is that part of your fuel qualification that, to check how sensitive you are to slightly off-normal in your --

MR. PETTI: We have not done it through testing. We've done some of that through analysis right now. And we see no change in predicted failure probability of the fuel over the specification range. You have to get down below say 20 microns. But our spec, we would not allow, we're allowing like 35 plus or minus three microns.

MEMBER ARMIJO: But so you're saying we didn't do the process qualification that way.

MR. PETTI: The problem is there are so many parameters. It gets so big.

MEMBER ARMIJO: So okay. I understand how you detect the defects in the silicon carbide, the leach, burn, leach. How do you detect either non-uniform or very thin silicon carbide? Somehow your process didn't --

MR. PETTI: We measure, we take a bunch of particles and we slice them and we measure the thicknesses and we make, you know, so we get a distribution.

MEMBER ARMIJO: Okay. It's by optical --

MR. PETTI: By optical metallography.

MEMBER ARMIJO: Got it.

MR. PETTI: And we also measure sphericity, that's the other thing I didn't say. And these are very, these are much more spherical particles then others use. So we think that's also very beneficial. So in terms of the accomplishments --

CHAIR BLEY: Is that different than the old pictures we saw of these that to me are rather irregular shaped particles?

MR. PETTI: Well partly you get faked out because of the magnification.

CHAIR BLEY: Of course.

MR. PETTI: But if you compare our fuel to the German fuel we are more spherical. It has to do with how we make the kernel actually compared to the rest of the world. We use something called an internal gelation process and they use external. And this goes back 25 years where the US stood behind that process and the rest of the world said no, we're going to go the other way. And now they're scratching their head when they represent stuff and they ask the question remember that argument?

We're seeing the results. So we have completed the most successful irradiation of fuel in the US to 19.4 percent FIMA. We have confirmed the expected superior radiation performance in UCO at high burnup. We have see in the PIE no kernel migration, no evidence of carbon monoxide attack of silicon carbide, no indication of silicon carbide attacked by the lanthanides.

Now if I take the fact that we saw zero failures out of 300 thousand and I do the 95 percent confidence estimate, that's below 10^{-5}. That is a factor of 20 better than the reactor requirement of two 10^{-4}. That plus the fact that this irradiation was much more severe in terms of temperature says that I think there's substantial margin.

AGR-2 is underway. No failures to date. Should complete by the end of this fiscal year.

MEMBER CORRADINI: Then I'm confused about your pink. What am I missing about the pink which I thought was --

MR. PETTI: The pink is gas release. Okay. This is failure effects. You have to take the release per failed particle, that's what collates the two.

MEMBER CORRADINI: Say it again slowly. I'm sorry.

MR. PETTI: That is gas release. This is failure fraction. And so when a particles fails there is a failure release fraction per particle that's a function of temperature that collates the two. And the number of particles in the --

MEMBER CORRADINI: So can I say it differently. So not to go back to the curve, but if I look at slide 20 the blue x's translates into a failure fraction of --

MR. PETTI: Less than 10^{-5}.

MEMBER CORRADINI: -- less than 10^{5}. And the pink ought to translate into something less than 10^{-4}?

MR. PETTI: Yes. So we think that there's margin here. In terms of AGR-3/4, it's underway. All the particles have failed. We're now kind of in a steady state mode. It started at this last Christmas. So we've go initial failure and then it kind of levels out. And so this data will be absolutely critical for us for the source term evaluations.

Now let me turn to PIE, which is the other sort of big new thing probably since the last time I talked. It took a lot of effort to get the infrastructure in place to do this work. PIE on this fuel is a lot different then pellet clad fuel given the size of the particles and the like.

We have three major objectives. A detailed characterization of the fuel after the irradiation in the reactor. A mass balance of the fission products for the source term so that you know that something is, has or has not been released. And then the high temperature safety testing to establish the fuel behavior under accident conditions.

And I show the two furnaces in Idaho and Oak Ridge that are used to do the safety testing.

And then some of the detailed techniques to handle these particles. We can deconsolidate the matrix and then retrieve the irradiated particles and do things with them at the particle level. It's an electrochemical process in an acid. So you slowly basically dissolve the matrix. It's not, it just.

MALE PARTICIPANT: I didn't know you could dissolve graphite.

MR. PETTI: Not dissolve, but that's not the right word. You can see in the picture there's shards, there's pieces of graphite. You just basically, it comes apart. And so we handle it like the eye doctor does the eye surgery. We have a big tv screen and plus on that in a hot cell. It's pretty amazing.

So the other thing that we're doing that's new is that we are basically throwing every technique that there is in material science at the fuel. The lab has received a number of real state of the art instruments to look at the nanoscale to do on irradiated material. And so the far right is a picture of a FIB a TEM sample of the silicon carbide layer.

We do the typical work with the SEM at the micron level. We also look above the micrographs. So we're getting the whole scale to look at everything. That black funny shaped object, that's a precipitate of palladium probably. So we're really looking at --

MEMBER CORRADINI: Where are you talking --

MR. PETTI: Nanometer, the nanometer. That's that right there. So we're looking at a different level than has ever been looked at the fuel. It's very exciting. We don't have answers for everything yet. But that's what we're doing to really try to understand what's going on in the fuel. So we expect that we're going to learn an awful lot more.

MEMBER ARMIJO: Are you studying the silicon carbide characteristics changes as a, with this technique?

MR. PETTI: Yes, we can do some work looking at irradiation effects as well. The other big thing that we've done is we've developed a methodology that if there is a defect that is contributing to the release that we're able to find it. We kind of call it the needle in the haystack.

So we start with 300 thousand particles in the capsule. We gamma scan the graphite that hold the compacts. There you see the picture, the three stacks. You look for hot spots. You do a tomography and there's a couple hot spots. Let's take that fuel and deconsolidate it and get the particles.

So actually that's where it goes from Idaho to Oak Ridge because they can look at every particle. We can't look at every particle, we can only look at only 60 to 100. They can look at all of them in the compact. And we put them in a machine called IMGA, which a basically a gamma spec. So every particle gets a gamma spec and you find them all with low cesium because that's what you saw, you saw some release.

Take that, put it in an X-ray thermograph. There's a 3-D reconstruction and we can find the defect and the nature of the defect. And in this case we know that this was a defect caused at laboratory scale that we're pretty sure that doesn't occur at industrial scale. But it gives us a heads up to okay, we better look at this on AGR-2 as well.

So never have we been able to go this level before. You'd see a release and you'd scratch your head after a furnace test and not understand it. We now today can go and find the particle that's defective. It takes a lot of effort. But we think,

a) it helps explain it, it takes the mystery out of why there was a release and just helps with the understanding. So this is a long process.

But we have basically found, we knew how many defective particles based on the quality control data we would expect to see. We have found them and we have characterized them.

MR. ALBERSTEIN: This is not to be confused with the earlier discussion about locating failed fuel in an operating reactor. That's a whole different --

MEMBER ARMIJO: That's a different thing.

MR. PETTI: It's hard enough to do --

(Crosstalk)

MEMBER ARMIJO: You're actually seeing the cracks in the silicon carbide?

MR. PETTI: Yes.

MEMBER ARMIJO: And on an individual particle, particles?

MR. PETTI: And that is a bad --

MEMBER ARMIJO: Might get some idea of what might have been the reason.

MR. PETTI: What caused it, right.

MEMBER ARMIJO: Compare a non spherical, at least those two.

MR. PETTI: Yes, well that's swelling. This is after irradiation and there's a crack in the buffer and so the kernel actually extrudes into the buffer.

MEMBER ARMIJO: It does.

MR. PETTI: Yes, from swelling.

CHAIR BLEY: Well plus look on here. You've got, is that the same?

MR. PETTI: Yes, it's the same one, yes.

CHAIR BLEY: Yes, because up here they've got it elongated this way.

MR. PETTI: They basically let through the cracks. The other thing this is, you know, you can't show it here. But this is now 3-D, which is also something that we've not been able to do. When you section, you know, you're taking your chances what do you find? 3-D we've got a completely new look at things which is very exciting to understand.

We know that this was caused from poor fluidization in the laboratory scale coater. The particle hit the wall and then there's soot on the wall and it causes an interruption in the silicon carbide layer. And if they're small it's not a problem. This one happened to be fairly large and it does cause a problem.

MEMBER ARMIJO: Very nice.

MR. PETTI: So there are a couple things now that we've learned from the PIE that are follow up items that the NRC has put in their assessment reports. And we wanted to give you a sense of what that is. It's, the questions probably two pages and the answers probably five pages. But I'm going to try to get you sort of snap shot.

First is that the irradiations in ATR don't produce enough plutonium. It has to do with the spectrum and the fact that we borated the capsules so we're absorbing some of the thermal neutrons in the boron. And then we don't get enough palladium because you get a lot more palladium from plutonium fission than you do uranium fission.

And we worry about palladium corrosion of the silicon carbide at high burnup. So the question is well how do you know that the test is representative? So we did a lot of calculations. And we found that what we get in AGR-1, about 40 percent below, for silver 40 percent below that expected in the reactor and palladium about 33 percent below that expected in the reactor at the peak burnup.

If you go to more average burnups the

numbers are smaller. So we went back and looked and the point for me that was the most convincing was that there was an experiment done back in the old days of pure plutonium fuel. So all fissions are in plutonium in Peach Bottom in a gas reactor for about a thousand days called FTE-13. Went to 70 percent FIMA. Typical temperatures and levels of damage.

Some palladium interaction was observed but no large scale degradation of the silicon carbide layer was observed by palladium. When you look at the volumetric concentration of palladium in those kernels it's about 75 times that in AGR-1. And the surface, if you say it's not volume it's surface, you'll go surface concentration, let's say around the kernel surface of a silicon carbide surface, it's 60x.

So we felt that the effect, the small 33 and 40 percent effects were small although the fact that there are tests out there with a lot more palladium. So we're continuing to look for palladium because the historical data suggests that this is something that should be a concern.

MEMBER ARMIJO: What's the mechanism by which palladium damages silicon carbide?

MR. PETTI: I think if it's in high

enough concentrations there are silicides on the base diagrams, eutectics low melting point.

(Crosstalk)

MR. PETTI: The palladium, yes. The palladium silicide. And there's a number of them on the phase diagram. But to date we have not seen any palladium attacked in AGR-1, which is actually amazing when you think about the times and the temperatures that we were operating at.

MEMBER CORRADINI: So you feel these others have bounded what you would expect to see as an effect?

MR. PETTI: I think that, yes. I think it's a couple things. I think accelerated irradiations, our historical database may be wrong because if they're all based on highly accelerated irradiations. Also it gets into the detail of the interface between the pyro carbon and the silicon carbide layers.

We have a different sort of interface then the historic US fuel. It's more German-like. And the Germans never saw this effect big. This was bigger in American fuel. So it's a combination of I think the, how we make the fuel and the fact that we used to test under very accelerated conditions.

MEMBER CORRADINI: So to ask, so this is your response to the RAI. And have you heard back from the staff on this relatively?

MR. PETTI: I think this remains a follow-up item. We just disagree. And we're going to, it's not like we're not going to keep looking. Of course we're going to keep looking in the PIE.

MEMBER CORRADINI: I understand. I just want to understand.

MR. PETTI: Right. But you'll hear from them I'm sure on it. So we wanted to give you sort of our perspective. The second one follow-up was they asked some questions on whether silver or palladium release, if you had a lot of it, would you degrade the silicon carbide? And there's some old theories, some old publications from the Germans that thought that, you know, if you could degrade the silicon carbide the cesium would come out.

And understandably you're worried about cesium. And then is there some sort of an enhancement under irradiation for cesium? So simply there's no evidence of this in the German database. Now under AGR-1 we released a lot of silver, 30, 40, 50, 60 percent in some compacts. One percent of the palladium is outside the silicon carbide in AGR-1.

So there are lots of these fission products have gone through.

Cesium is low. There is no release of cesium. If you see cesium, it means there's a defective particle because when we do the measurement we'll see the uranium as well. So the silicon carbide seems to be very, very good and not susceptible to this theory of silver and palladium degrading the silicon carbide, we just don't see any data yet.

So we're seeing minimal release of cesium in the matrix. And if there were palladium degradation you should see a lot more cesium. I mean we're not seeing anything, not even one particle's worth.

MEMBER CORRADINI: So can I ask you a different question?

MR. PETTI: Yes.

MEMBER CORRADINI: So in the original TRISO particle you've got the kernel, the silicon, no I guess you have a kernel, a buffer layer and then the silicon carbide and another buffer layer.

MR. PETTI: No, no, no. A kernel, a buffer, an inner pyro carbon layer, a silicon carbide and an outer pyro carbon layer.

MEMBER CORRADINI: Thank you. So which is duplicative that could be lost? In other words if tomorrow you were to say I want to simplify the process. You've made it clear that silicon carbide is important. So what would you take off if you put this in a binder or compact? The outer pyrolated carbon layer? This is an off the wall sort of question.

MR. PETTI: They're all there for a different, they're each there for a reason.

MEMBER CORRADINI: So they're all critical, nothing is removable?

MR. ALBERSTEIN: Necessary and sufficient.

MR. PETTI: Yes, the inner layer is there because the chemicals that are used to make silicon carbide, chlorine can attack the kernel so the pyro carbon's there to protect the kernel. The outer pyro carbon is there so that the matrix has something to grab to. And I think it would be harder to grab to pure silicon carbide. That's what I would get rid of.

MR. ALBERSTEIN: It also keeps the silicon carbide in compression under irradiation.

MR. PETTI: But I do not believe that was designed that way. I think that was a, look at that. The big benefit is that the pyro carbon's shrink and keep that silicon carbide in compression.

MEMBER CORRADINI: So what you're telling me, I'm a little bit off base. But let me ask and then you'll tell me go away, we'll talk later. So what you're telling me is you have a recipe. The recipe works. Don't screw with the recipe because it works.

And the understanding of taking something off has an effect, I'm most interested in the pyrolated carbon layers, not the buffer, not the, because you've made it very clear what the SiC does. So if I took out something could you predict the effect or you'd have to test the effect?

MR. PETTI: It depends on what it is. If it's in the particle, I can predict the effect. I can tell you what happens when you lose the outer pyrolated carbon.

MEMBER CORRADINI: You could?

MR. PETTI: Yes, but if you want to put a different matrix or something that's different.

MEMBER CORRADINI: No, I have a question, something came to my head.

MR. PETTI: Right. But the nature of how

the outer pyro carbon adheres to the matrix, you know, they're both carbon so it works. So, you know.

MEMBER REMPE: Wasn't there an example with the NPR where they added another layer and that didn't work so well too?

MR. PETTI: Yes, the program is very, sort of cautious about just trying stuff.

MEMBER CORRADINI: So let ask the final off-the-wall question. So if the pyrolated carbon layers were reduced in size, is this the minimum thickness or is this just from a recipe standpoint the acceptable thickness?

MR. PETTI: You might be able, particularly on the outer you might be able to go thinner.

MEMBER CORRADINI: Fine, okay, I'll stop.

MR. PETTI: I think so. The inner --

MEMBER CORRADINI: You and I had talked about other things before so I'm --

MR. PETTI: Right but he's thinking, he's in a completely different sphere.

MEMBER CORRADINI: It was just a question.

MR. PETTI: The other thing is that if you look at the cesium release that we can measure in the matrix, it's very consistent with the diffusion coefficients that we got from the Germans. So that is conservative. So we'll continue to look at this. But right now we see no evidence of any degradation. Although a priori one might think with all of this, this is one of the great surprises of AGR-1 that we've moved a lot of this material outside of silicon carbide and it just looks absolutely fine.

So okay. So --

MEMBER CORRADINI: Do you know why?

MR. PETTI: We're getting there. But we're in the middle of PIE. So I think we're really trying to get to a new level of understanding of the performance and the transport. In terms of the mass balance we're looking at silver, cesium, strontium, europium, celium, palladium. We do look for ruthenium, Tom, but we just don't see it.

We're characterizing the microstructures at both the micro and the nanoscale. No palladium corrosion or attack has been observed. But the models would predict we should have seen significant amount. And they conservatively, the models I can't show you all this today given the time, conservatively overpredict how much cesium you get under normal operation.

DR. KRESS: Does that worry you?

MR. PETTI: No, in the end I think this is because if you look at the old German silicon carbide, you can see little bits of porosity.

DR. KRESS: You know, when I see something like that I worry that my model's not correct. You know, it's a concern.

MR. PETTI: Yes, no, I think it just has to do with the microstructure is just so much better material. That's what I think.

So we've done accident heatup testing. This is the real test of the fuel where you saw the plot on the right. We basically put the fuel in the furnace for hundreds of hours at 16, 17 and 18 hundred. We've completed five tests at 16 and 17. And we're going to do 1800 this year.

We're actually going to do one that mimics that blue curve, a time temperature curve instead of a constant because there's concerns about that. That's on the plate for this year. And we're seeing very, very low release. What we're seeing released is in fact material that diffused into the matrix under irradiation. There is no release from intact particles. This is absolutely stunning result.

The silicon carbide is this good. So to test that because you're really looking at small numbers is we will deconsolidate and test a bunch of particles and just show that's the case. So this is very, very interesting in terms of the safety testing. And here's my one plot where I compare now the releases in these high temperature heatups against what a diffusion model would predict taking our best estimate diffusion coefficients.

So all the ones with data points are the experiments. And the blue line is the prediction of just what you'd expect from pure diffusion through the particles. And you can see that we overpredict, particularly in the strontium. The purple line is a defect, there's one defective particle. If you look at the release of cesium, the purple line and strontium the purple line, they're the same. They're 2.4 10^{-4}. They're roughly one particles inventory.

So we then said well we can do that. So we shut off all the diffusion in the model and we've calculated one defective particle and that's the solid green. And you can see that we underpredict. And so that's not a huge surprise. These are all UO2 correlations so it looks like maybe UCO releases cesium a little bit greater. And the strontium the chemistry is very different in UCO. It doesn't come out of UO2 but we do expect that there could be a little mobile in UCO.

So this is a snap shot. But will the real, the reactor vendor codes sort of predict are these blue lines.

MEMBER CORRADINI: Remind me. PARFUME is going to be what you would recommend to the owner, operator as their fuel?

MR. PETTI: Not necessarily. I really don't know what --

MEMBER CORRADINI: The PARFUME is the currently accepted way to do an estimate. And since I don't remember, you told us this two years ago in a meeting and I can't remember.

MR. PETTI: This is, yes, I mean the vendor's going to do what the vendor's going to do. This is perhaps the most complex, the most detailed code. But it may be impractical to do something like this at reactor scale because it's so detailed.

MEMBER CORRADINI: Okay.

MR. PETTI: But again, so there's lots of margin it looks like because of course we're not releasing anything. So in terms of our accomplishments we're nearing completion on the safety testing for AGR-1. We'll finish that up this fiscal year demonstrating real robustness. Again, very low releases after hundreds of hours, no particle failures and no noble gas release measured, which means you have not failed all the layers.

So we do see, when we see cesium it's usually indicative of a defect. We go look, we find it. It's a defect. Again, the releases are associated with fission products that have diffused into the matrix. No diffusive released from intact particles is what we believe is going on right now.

So in terms of that failure fraction specification it's six times 10^{-4}. We've not tested enough fuel to make a statistical statement. But I believe if we test enough, we may end up with a zero out of how ever many we test and what that means statistically. So we think that we'll be able to do that.

We still need the data on the water and air ingress. That's in the plan. And the historical database on the diffusion coefficients seemed to overpredict the measured releases. And I think that's largely because the silicon carbide is just better than what the Germans made.

CHAIR BLEY: At one point, I think it was talking about kernel migration, Tom had asked you what, how much carbide you need. And you said it wasn't real precise. There are other characteristics that are effected by the carbide, some of which you just showed us. Is there range or did you just pick one and it's worked really well or have you got, there's a minimum you need or is there a maximum that gets you in trouble with cesium?

MR. PETTI: Yes, so there's a, I mean what you do is you look at the thermochemistry and if you put in too much carbon then too many of the lanthanides get mobile. If you don't put enough you don't tie it up, the chemistry doesn't work. And so I think our spec is like 20 to 40 percent and we just hit --

CHAIR BLEY: Okay. So it's fairly broad, but it's --

MR. PETTI: Fairly broad. But and it depends on the burnup you want to go to. So in the old days in the NPR where we had HEU fuel you changed the spec a little bit to expect the higher burnup and the greater number of fissions. So, yes, this was all developed, you know, 25 years ago. It's just taken us this long to prove what was very compelling on paper for thermodynamics that it works.

So we'll complete our safety testing in PIE this year including the safety testing at 1800. And as Don knows at the last gas reactor conference, 1800 is when the Germans began to see degradation. I'm not convinced. We did 1700 and saw no degradation. We may not see degradation at 1800, which again would be very encouraging.

We'll complete AGR-2 this fiscal year, complete AGR-3/4 in 2014. And then do the follow on safety testing in PIE in 2014 and 2015. AGR-5/6/7, the qualification and margin testing is scheduled for 2016. The follow on PIE campaign 2018 to 2020. And then AGR-8 follows beyond that.

So in one slide sort of what's the key results in terms of fabrication. I think we understand the process much better. We brought today's technology to it. We've improved fabrication and characterization by the vendor. We've had outstanding irradiation performance of a large, statistically significant population under high burnup, high temperature HTGR conditions. We've confirmed the expected superior radiation performance of UCO at the high burnup.

The PIE indicates a lot of silver release because we ran the experiment very hot. But it's consistent with the models. They can calculate silver reasonably well. No cesium released from intact particles under irradiation. No palladium attack or corrosion despite large amounts of palladium outside the silicon carbide.

The initial safety testing demonstrates robustness of UCO TRISO under the depressurized conduction cooldown condition. Low release in the intact particles. All the releases to date attributed to defects or fission products that were released under irradiation and moved in the furnace.

MEMBER CORRADINI: The last bullet, why are you saying under or depressurize, excuse me? I misread that.

MR. PETTI: And then no failures to date. So we are learning some very new things in terms of what we think the real limits of UCO TRISO are. Today we don't really know. So there's a lot we're going to learn here. So in summary, we're providing the data necessary to understand the behavior for the modular HTGR.

We're laying the technical foundation needed qualify the fuel made to process product specifications within the envelope of the operating and accident conditions that bound modular HTGRs.

Our results to date are consistent with the current design assumptions that are being made. And we're obtaining additional data to support the development and validation of models.

And our results to date are generally consistent with the safety design basis, including the functional containment and mechanistic source term approaches presented today. So it is all sort of lining up and fitting together.

CHAIR BLEY: Very good. Any more questions from the Committee? And that was the last item on our agenda.

MS. BANERJEE: Public, are we having --

CHAIR BLEY: Well I was going to do that now. But if we can open the phone line. But I'll ask inside now.

FEMALE PARTICIPANT: I can go and check.

CHAIR BLEY: Okay. Are there any comments from any members of the public here in the room?

MR. SHAHROKHI: I'd like to make a short statement.

CHAIR BLEY: Please come to the mike and identify yourself.

MR. SHAHROKHI: My name is Farshid

Shahrokhi. I work for AREVA US in Lynchburg, Virginia. Today I'm representing the NGNP industry alliance. The alliance is an organization, it's a 501(c), a nonprofit organization. Our current members are about 13 or 14 members. It includes end users, operators and designers and suppliers.

And we've been involved with the NGNP project. In fact we've just signed a public, part private partnership with DOE. As our first task is to perform an economic analysis and some trait studies on our selected design which is the AREVAs steam cycle high-temperature gas-cooled reactor. It's a 625 megawatt prismatic design, two-loop, very similar to some of the pictures that you've seen.

It's a larger, higher power reactor, but it is a prismatic design. You use the compacts fuel. We've been involved with the NGNP project since its inception. Some of our members have been supporting the NGNP project. And we are hoping that the, we're closely following the interaction, these, this generic licensing interaction.

And we have put together a business plan and we're trying to capitalize the development venture of the business plan which says we are talking with investors to get us going to begin the design of this business plant. No one company or can afford to design this reactor or bring it into a commercial role so we are looking for investors. And the investors and member, organization members are looking for clarity, licensing clarity and continuation of the fuel qualification which is really at the crux of this technology. Thank you.

CHAIR BLEY: Thank you. And --

MS. BANERJEE: Theron is opening the line up for public comment. So you can ask again.

CHAIR BLEY: Any other comments in the room? We think we've opened the phone line. Could somebody out there just say you're on the line?

MS. BANERJEE: There were four people on the line to start with.

CHAIR BLEY: We were expecting, well there were four on the line earlier.

MALE PARTICIPANT: I'm on the line.

CHAIR BLEY: Good, thank you. Does anyone on the line care to make a comment? Hearing none we'll end with the comments. I'd like to go around to the Members and hear anything you have to say. I'll start with Mike.

MEMBER CORRADINI: I think, I appreciate Idaho and the contractors for the DOE and their presentation. I think it was a good review since we haven't heard from them I think now in a couple years, at least in this venue. Some of us have heard from them.

Yes, I guess I'd only say that I'd look forward to see how the staff's, as Don was saying, what the staff's roll-up of comments or their assessment in how to end this off.

I think it's important that if there's going to be some delay in activities within the staff it's important that this is wrapped up in some way that we get a clear idea where the staff sits on a lot of these issues so that when it's picked up again we don't have to revisit any of these things. And to me that's very important otherwise we're going to lose the certainty or at least some certainty as to where to go from here relative to this, to the NGNP advanced reactor. But other than that I would just thank the INL and their staff.

CHAIR BLEY: Thanks, Mike. Charlie.

MEMBER BROWN: As an electrical puke I was just in a learning experience trying to figure out what's been going on for the last, in this program. So this was a nice summary today. I did appreciate the detail that was presented. I even understood some of it. Don't ask me which part because I probably couldn't repeat it.

CHAIR BLEY: And, Tom, I know you'll give us written comments. But I'd appreciate anything you want to say now.

DR. KRESS: We'll give you written comments. Maybe I ought to confess my bias. I've always been an admirer of this gas-cooled reactor concept. I think their approach to licensing is very good. I like it very much. I like their fuel concept and how they're making them.

I have a little bit of concern about the, whether you can really show the quality. But put some of those concerns to bed. I'm still glad you have a monitoring system in the primary system just in case. That's kind of one of my ideas of a defense in depth. You know, when we were talking about defense in depth.

CHAIR BLEY: Good example.

DR. KRESS: I personally don't think you need any containment. So I think the staff has done a real good job addressing all these questions. And I like some of their questions and I like some of their positions. One thing I think I had a little concern with was you're obviously going to make the

QHOs. I mean I can tell you that without even calculating them.

But most of the PRAs for LWRs also calculate the land contamination, the cancers and effects of that nature. If you are going to release some fission products and they're going to go beyond the EAB and they've got things in them like the linear no-threshold. And I don't know, I know that there's no acceptance criteria, no regulations in there.

But I think you ought to think about those things if you're going to, somebody's going to ask you about them somewhere along the line. If they don't, I'm going to ask you. But I think that's the one area I think you need to show that you have an acceptable thing. But other than that I like everything I've heard so far. I'm glad to hear the update.

MR. ALBERSTEIN: I think one of the people that might be the first to ask that question would be the owner of the co-located process heat using facility.

DR. KRESS: That might very well be. But other than that I have better comments when I write them down.

CHAIR BLEY: Okay. Thanks, Tom. Charlie, you get another shot.

MEMBER BROWN: Yes, I just one for the uninitiated for myself in listening to the defense in depth aspects, that's the part I would be interested in. I'm not quite as sanguine about the validity of this just so super particle that will never break and release nothing that you need no other more passive, which is in a very active, hot environment and without some type of passive, non, in other words a blacksmith type technology containment of some sort.

And if you don't need a high pressure containment under the circumstance, but a sealed containment. So I'm just, I'll be interested in hearing the justification in more detail on that. But there's, I'm not as enthusiastic about that thought process as a couple of the comments have indicated today, so.

CHAIR BLEY: Thank you.

MS. BANERJEE: That's one action item kind of thing we have from this meeting for --

MEMBER BROWN: Yes, I understand that, that you noted that one. So I appreciated that one being laid on the table. Excuse me, Dennis, I'm sorry.

CHAIR BLEY: That's all right. You have a free hand here. I'd like to thank everybody as well. It's been a very good day. I don't know how you pack so much into the little time we allowed you. But you did a bang up job of it. It was nice to hear what's happened since the last briefing I had on the experiments and the like.

And it's all very impressive and covers a lot of the things I've worried about. I think that this idea of coming back and talking about defense in depth is probably very useful. Staff here has tried several times to define defense in depth. It's out there in many forms, many ways, many people have tried. There are several documents have been prepared over the last 20 years dealing with that. They all come at it a little differently.

To me anything you do to lower the likelihood of release or to control the amount of release is beyond what you absolutely need is some form of defense in depth. And certainly something to cover the uncertainty aspects. But the idea that the only thing that's defense in depth is a pure physical barrier, well they aren't pure and they aren't perfect. So you have problems with those as well. So looking at the wide range of things that can do it would be very interesting.

In any case, thank you very much again. I appreciate you all coming and your answers to all the questions. And at this point we'll adjourn the meeting.

(Whereupon, the hearing in the above-entitled matter was concluded at 2:17 p.m.)

U.S. DEPARTMENT OF ENERGY | Nuclear Energy

ACRS Future Plant Designs Subcommittee Meeting

NGNP Introduction

Carl J. Sink, Program Manager
Office of Nuclear Energy
U.S. Department of Energy

January 17, 2013

U.S. DEPARTMENT OF ENERGY

Nuclear Energy

NRC- DOE Licensing Strategy – 2008 (Report to Congress)

- **“It will be necessary to resolve the following NRC licensing technical, policy, and programmatic issues and obtain Commission decisions on these matters”**
  - Acceptable basis for event-specific mechanistic source term calculation, including the siting source term
  - Approach for using frequency and consequence to select licensing-basis events
  - Allowable dose consequences for the licensing-basis event categories
  - Requirements and criteria for functional performance of the NGNP containment as a radiological barrier

NRC- DOE Licensing Strategy – 2008 (Report to Congress)

- The best approach to establish the licensing and safety basis for the NGNP will be to develop a risk-informed and performance-based technical approach that adapts existing NRC LWR technical licensing requirements in establishing NGNP design-specific technical licensing requirements.

- This approach uses deterministic engineering judgment and analysis, complemented by probabilistic risk assessment (PRA) information and insights, to establish the NGNP licensing basis and requirements.

Continued DOE Focus on Licensing Framework

Secretary Chu letter to Congress in October, 2011 reinforces the priority that DOE places on establishing the HTGR licensing framework, based on the related NEAC recommendation

- "The NEAC also recommends that the Department continue research and development, as well as interactions with the Nuclear Regulatory Commission, to develop a licensing framework for high temperature gas-cooled reactors."

Licensing Framework Interactions with NRC

White Paper	Submittal Date	NRC Public Meeting(s)
NGNP Defense-in-Depth Approach INL/EXT-09-17139	December 9, 2009	March 8, 2010
NGNP Fuel Qualification White Paper INL/EXT-10-18610	July 21, 2010	September 2, 2010 October 19, 2011 April 17, 2012 July 24, 2012 September 20, 2012 November 14, 2012
HTGR Mechanistic Source Terms White Paper INL/EXT-10-17997	July 21, 2010	September 2, 2010 October 19, 2011 April 17, 2012 July 24, 2012 September 20, 2012 November 14, 2012

Licensing Framework Interactions with NRC – cont

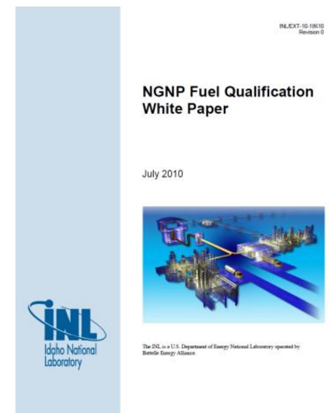
White Paper	Submittal Date	NRC Public Meeting(s)
NGNP Licensing Basis Event Selection White Paper INL/EXT-10-19521	September 16, 2010	November 2, 2010 April 16, 2012 May 16, 2012 July 10, 2012 August 22, 2012 September 19, 2012 November 14, 2012
NGNP Structures, Systems, and Components Safety Classification White Paper INL/EXT-10-19509	September 21, 2010	November 2, 2010 July 10, 2012 September 6, 2012
Determining the Appropriate EPZ Size and Emergency Planning Attributes for an HTGR INL/MIS-10-19799	October 28, 2010	January 26, 2011 November 14, 2012

Licensing Framework Interactions with NRC – cont

White Paper	Submittal Date	NRC Public Meeting(s)
NGNP Probabilistic Risk Assessment White Paper INL/EXT-11-21270	September 20, 2011	April 12, 2012 September 19, 2012
Modular HTGR Safety Basis and Approach INL/EXT-11-22708 (submitted for information only)	September 6, 2011	None

Licensing Framework Interactions with NRC – cont

To supplement the above public meeting interactions, NGNP has also provided written responses to approximately 450 NRC Requests for Additional Information focused primarily on the topics of licensing basis event selection, mechanistic source terms, and particle fuel qualification



<u>**NRC Document**</u>	<u>**Transmittal Date**</u>
NGNP – Assessments of White Papers on:	February 15, 2012

- *Fuel Qualification and Mechanistic Source Terms*
- *Defense-In-Depth Approach, Licensing Basis Event Selection, and Safety Classification of Systems, Structures, and Components*

NRC Staff Positions Requested by DOE

- **NGNP transmitted a letter to NRC on July 6, 2012 reinforcing areas of priority for licensing framework development**
  - Consistent with focus areas summarized in NRC to DOE letter from February 15, 2012

- **NRC staff positions have been requested in four key areas**
  - Functional Containment Performance Requirements
  - Licensing Basis Event Selection
  - Establishing Mechanistic Source Terms
  - Development of Emergency Planning and Emergency Planning Zone Distances

Purpose of Today's Meeting

- **Presentation topics in support of licensing framework**
  - HTGR Safety Design Bases
  - Licensing Basis Event Selection Process
  - Functional Containment Performance and Mechanistic Source Terms
  - Siting Source Terms
  - Fuel Qualification and Radionuclide Retention
- **DOE is focused on the resolution of long-standing HTGR "licensability" issues, and establishment of key parts of the licensing framework**
- **Eliminate the prevailing cloud of uncertainty surrounding these issues that is challenging both DOE and the private sector regarding NGNP deployment**

HTGR Safety Approach and Design Basis

ACRS Future Plant Designs Subcommittee Meeting

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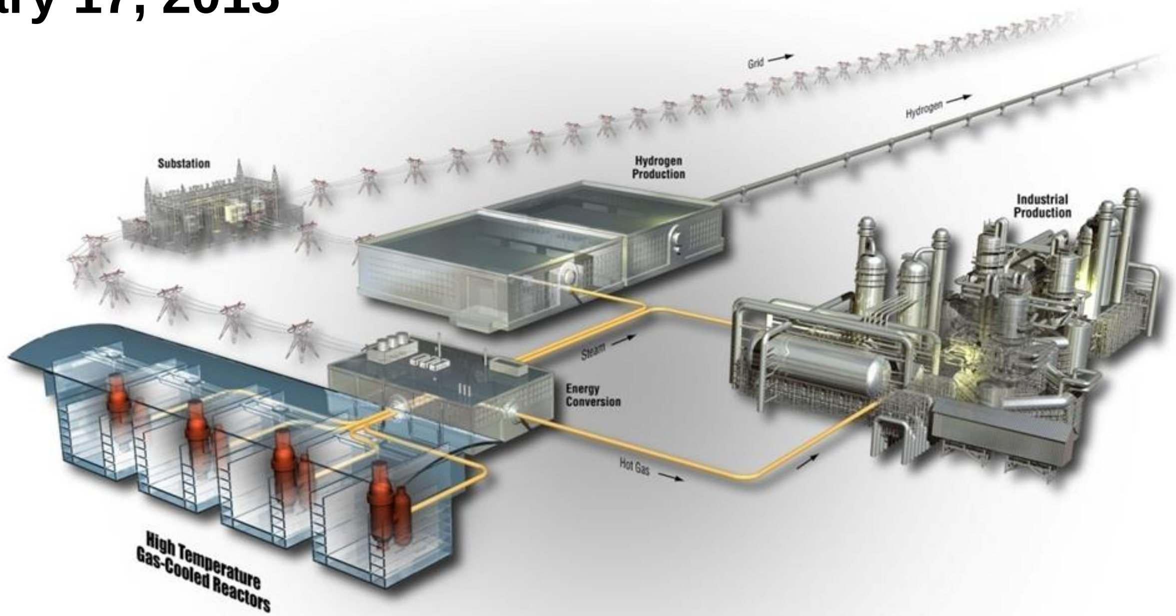

NGNP Next Generation Nuclear Plant

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Meeting Agenda

- HTGR Safety Design Bases
- Licensing Basis Event (LBE) Selection Process
- Functional Containment Performance and Mechanistic Source Terms
- Siting Source Terms
- Fuel Qualification and Radionuclide Retention

Modular HTGR Safety Design Objective

- Do not disturb the normal day-to-day activities of the public
  - Meet EPA Protective Action Guides at the plant boundary (EAB) for event sequences with a frequency greater than or equal to 5×10^{-7} per plant year

Modular HTGR Safety Design Approach

- Utilize inherent material properties
  - Helium coolant – neutronically transparent, chemically inert, low heat capacity, single phase
  - Ceramic coated fuel – high temp capability, high radionuclide retention
  - Graphite moderator – high temp stability, large heat capacity, long response times
- Develop simple modular reactor design with passive safety
  - Retain radionuclides at their source within the fuel
  - Configure and size reactor for passive core heat removal from reactor vessel with or without forced or natural circulation of pressurized or depressurized helium primary coolant
  - Large negative temperature coefficient for intrinsic reactor shutdown
  - No reliance on AC-power
  - No reliance on operator action and insensitive to incorrect operator actions

Multiple Barriers to Radionuclide Release

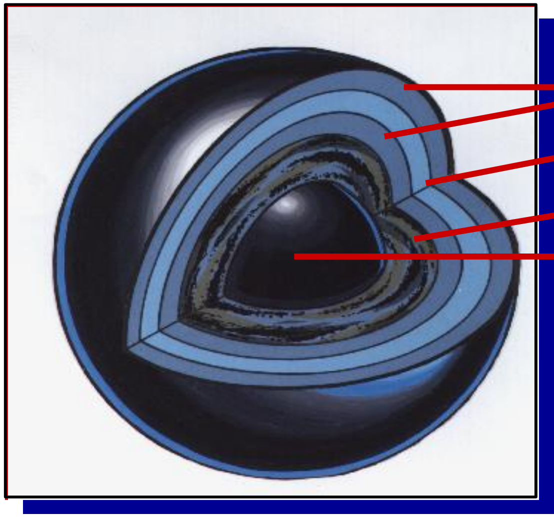
- Fuel Kernel
- Fuel Particle Coatings (most important barrier)
- Compact Matrix/Graphite

Fuel Element

- Helium Pressure Boundary

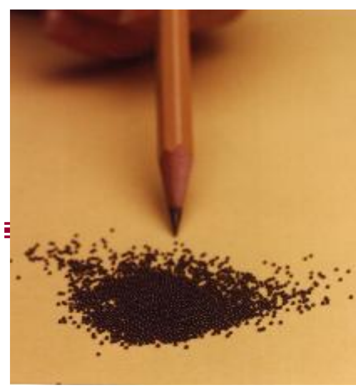
- Reactor Building

HTGR Fuel



Pyrolytic Carbon (Inner and Outer)
Silicon Carbide
Porous Carbon Buffer
Fuel Kernel

TRISO coated fuel particles (left) are formed into fuel compacts (center) and inserted into graphite fuel elements (right)



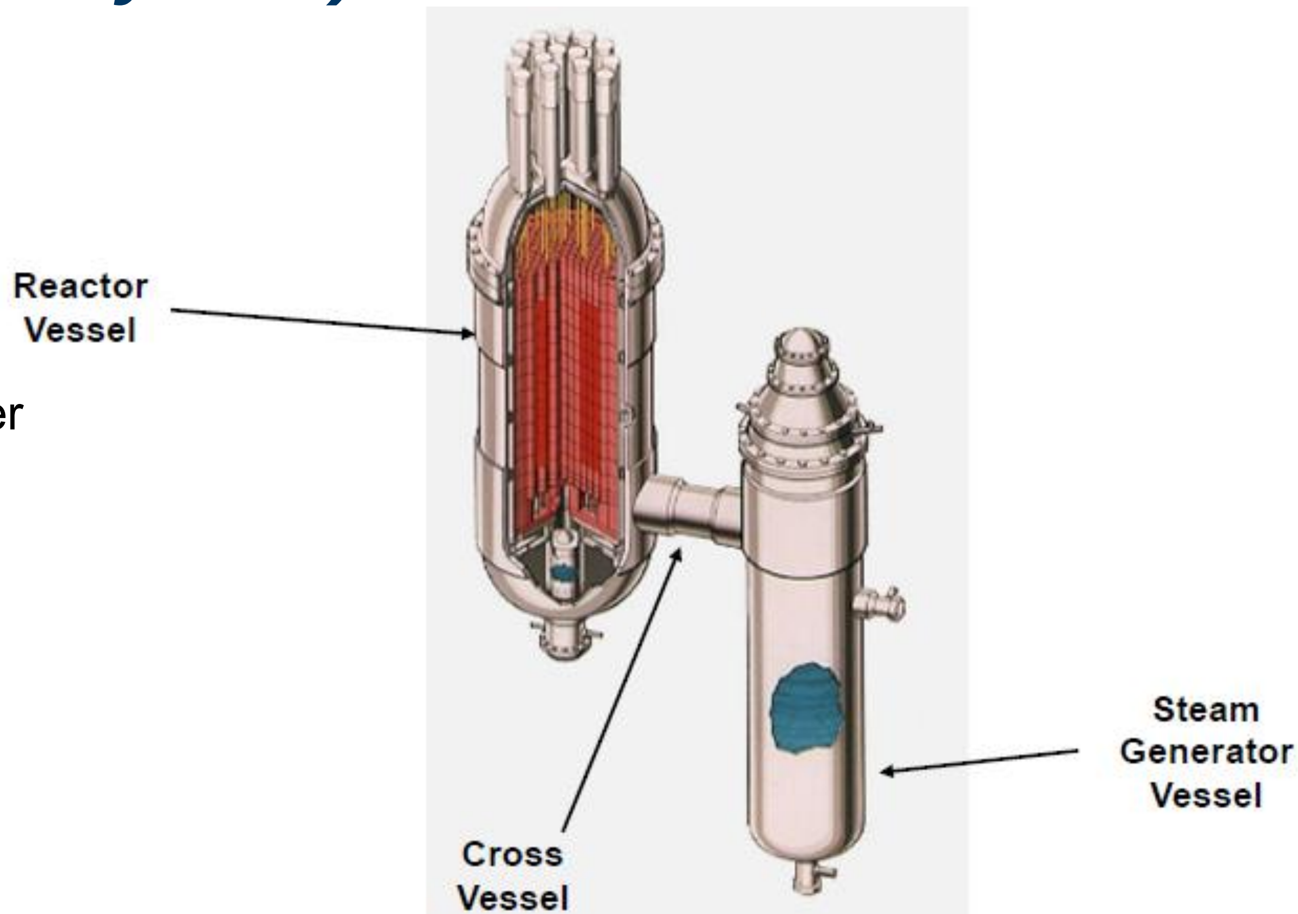
Particles

Compacts

Fuel Elements

Helium Pressure Boundary (HPB) (MHTGR Vessel System)

- ASME B&PV Code Section III pressure vessels
- Higher pressure colder helium in contact with vessels
- Loss of helium pressure does not cause loss of cooling



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Reference MHTGR Embedded Reactor Building

Protects pressure vessels and RCCS from external hazards, provides additional radionuclide retention, limits air ingress following HPB depressurization

Multi-cell, reinforced concrete

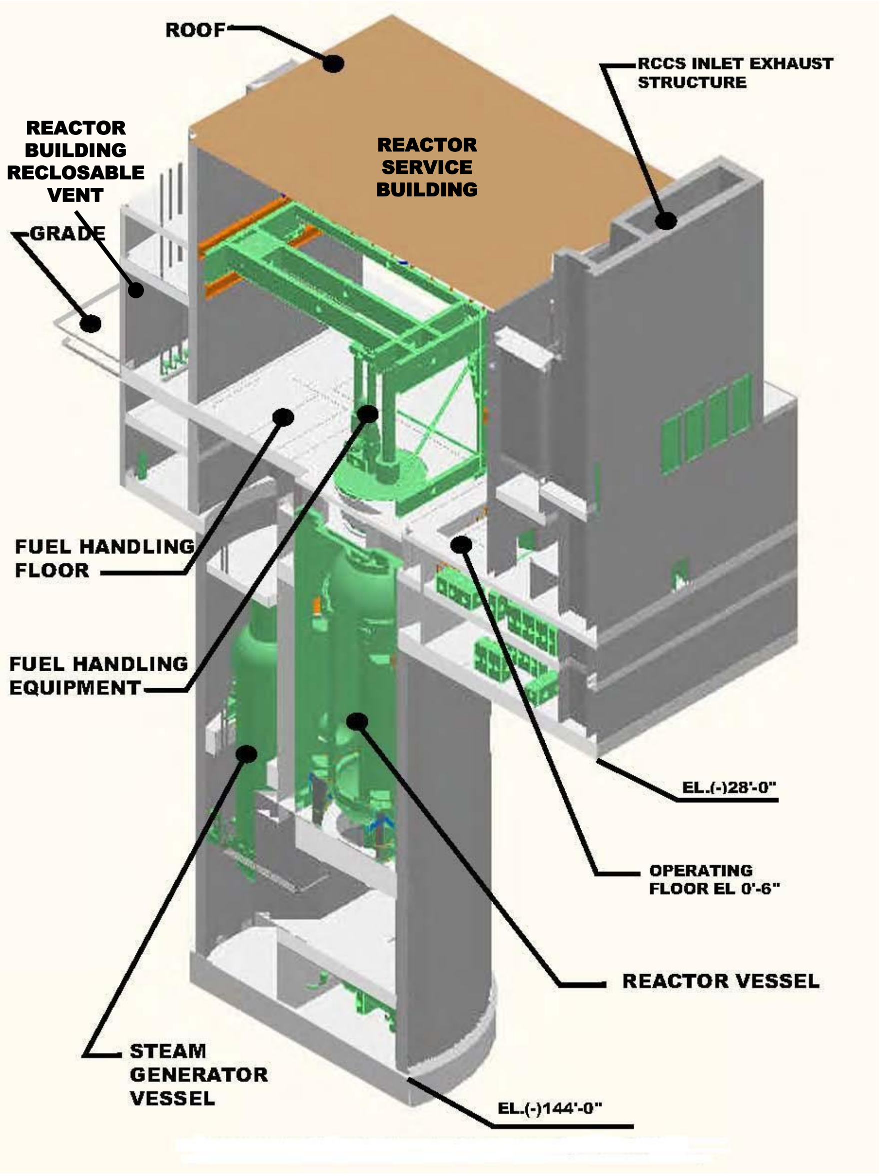
Seismic Category I

External walls ~ 3 ft thick

5 ft slab between RV and SGV cavities

Slab at grade provides
- Biological shielding
- Missile protection
- Plugs for equipment access
- Control for personnel access

Moderate Leak Rate (100% per day)



Responsive to Advanced Reactor Policy

- Use of inherent or passive means of reactor shutdown and heat removal
- Longer time constants
- Simplified safety systems which reduce required operator actions
- Minimize the potential for severe accidents and their consequences
- Safety-system independence from balance of plant
- Incorporate defense-in-depth philosophy by maintaining multiple barriers against radiation release and by reducing the potential for consequences of severe accidents
- Citation of existing technology or which can be satisfactorily established by commitment to a suitable technology development program

Key Element of Safety Philosophy

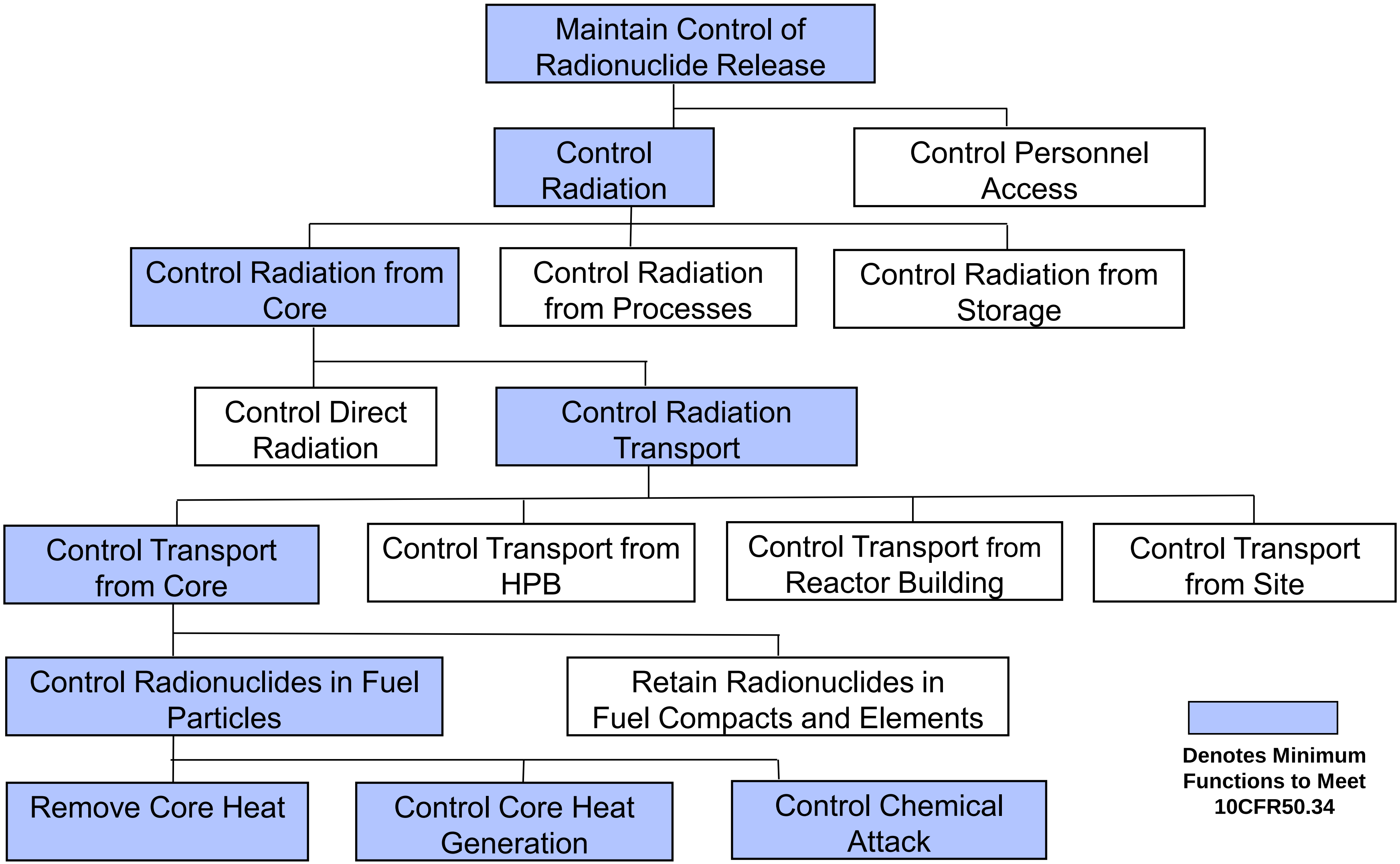
- Emphasis on retention of radionuclides at source within fuel means:
  - Manufacturing process must lead to high quality fuel
  - Normal operation fuel performance must limit potential for immediate radionuclide release during off-normal conditions – coolant is continuously monitored during operation
  - Off-normal fuel performance must limit potential for delayed radionuclide release to a small fraction of non-intact fuel particles from manufacturing and normal operation conditions

Safety Design and Technology Development Focus

- High fuel manufacturing quality and normal operation fuel performance ensure that modular HTGR could release activity outside of fuel barriers (e.g., circulating within HPB) and stay within offsite accident dose limits
- Thus, safety design and technology development focus is on limiting incremental releases from fuel during off-normal events
- Promising AGR fuel development program results to date

Next Generation Nuclear Plant

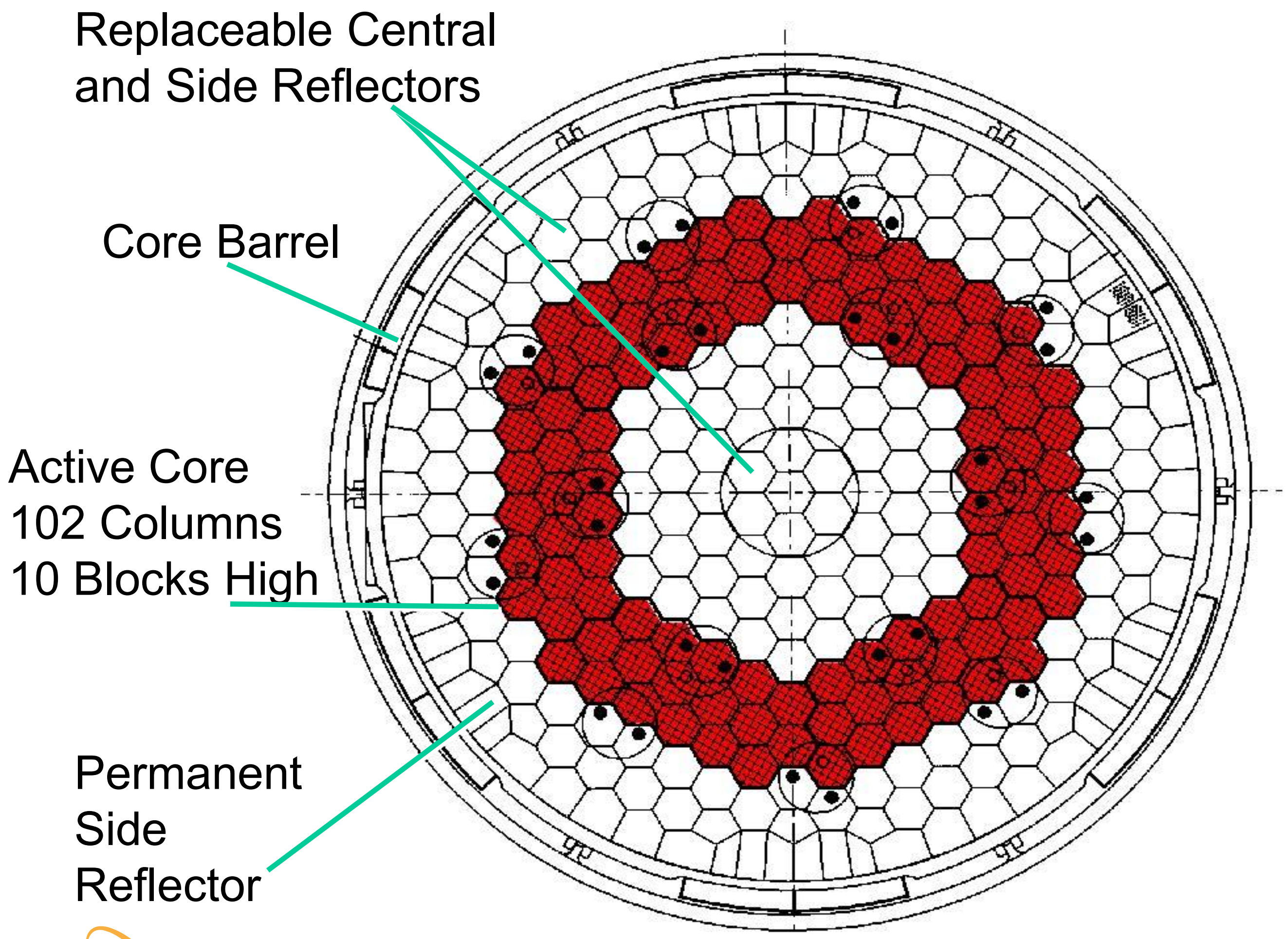
Functions for Control of Radionuclide Release



Removal of Core Heat Accomplished by Passive Safety Features

- Small thermal rating/low core power density
  - Limits amount of decay heat
  - Low linear heat rate
- Core geometry
  - Long, slender or annular cylindrical geometry
  - Heat removal by passive conduction and radiation
  - High heat capacity graphite
  - Slow heat up of massive graphite core
- Uninsulated Reactor Vessel (RV)
- Reactor Cavity Cooling System (RCCS)
  - Natural convection of air or water

Annular Core Optimizes Passive Heat Removal



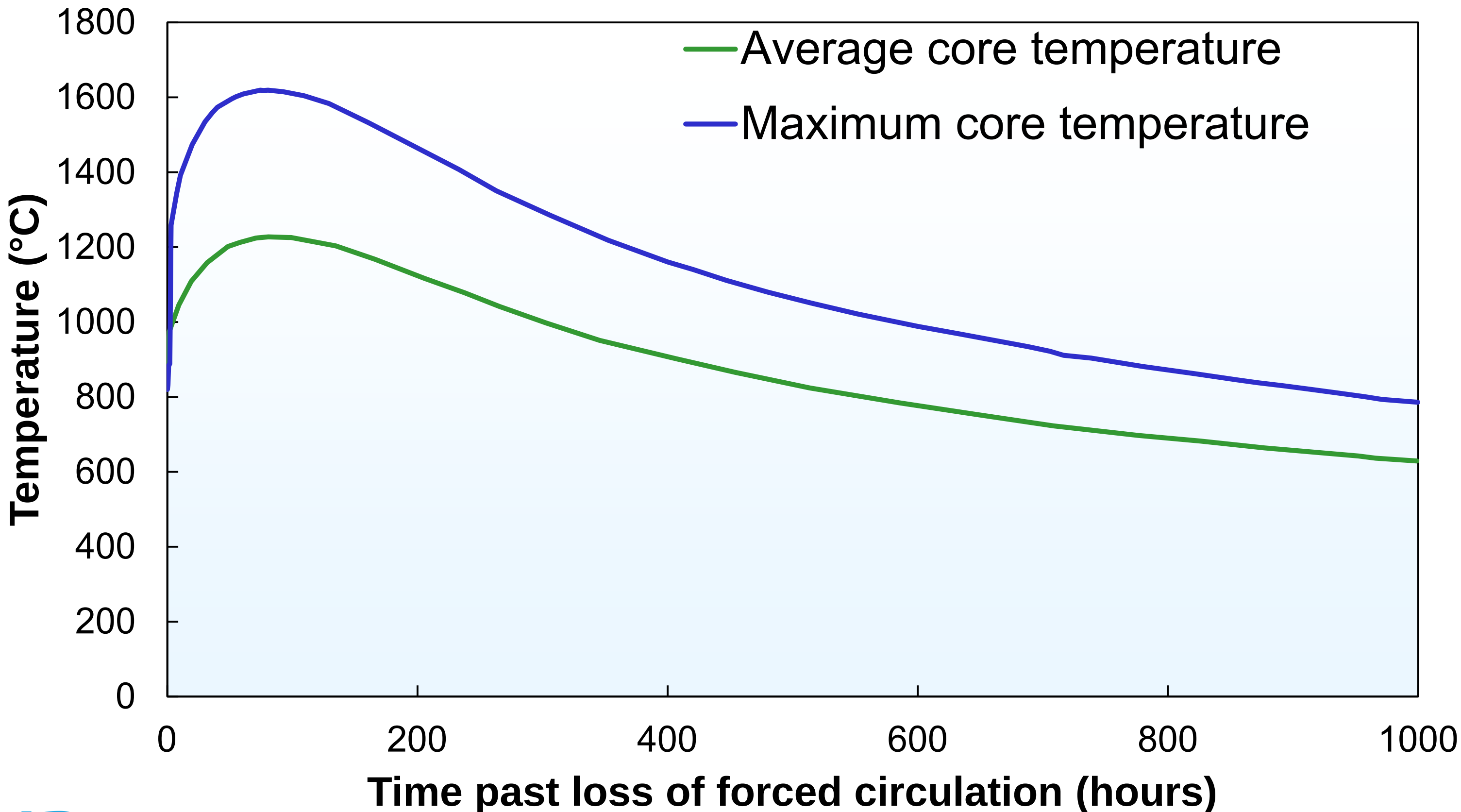
Modular HTGR utilizes annular core geometry to:

1) shorten conduction path
2) enhance surface to volume ratio

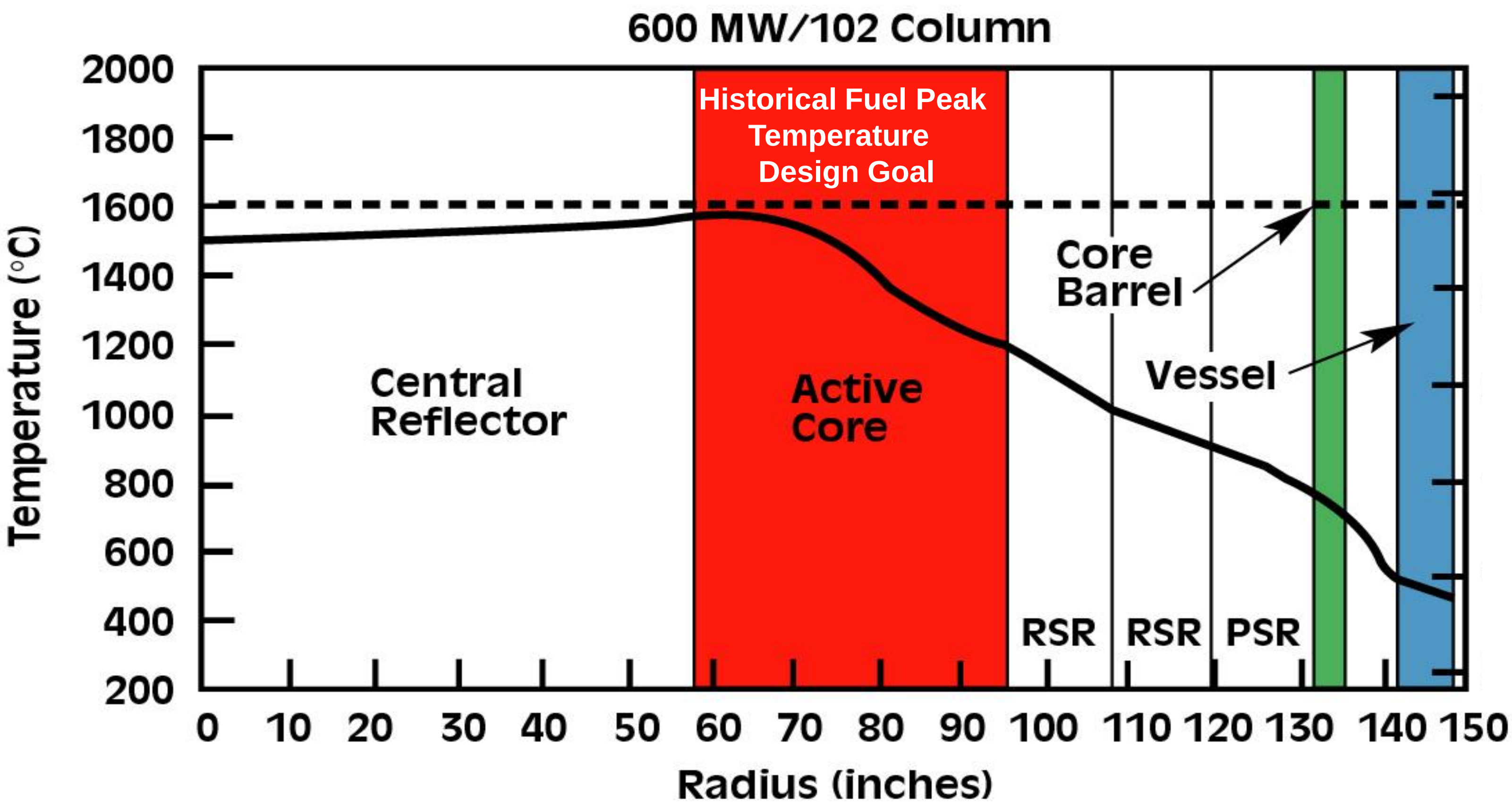
Depressurized Loss of Forced Cooling Events (DLOFCs) Demonstrate Passive Heat Removal

- DLOFCs are rare events in which the helium coolant is depressurized and in which the two independent forced cooling systems are both immediately and indefinitely unavailable to remove core heat
- Consequently, the core gradually heats up and the heat is removed by conduction, radiation, and convection radially to the RV to the RCCS
- DLOFCs consist of three phases that can overlap depending on the size of the leak/break in the HPB:
  - Initial depressurization (minutes to days)
  - Subsequent core heatup (~2 to 4 days)
  - Subsequent core cooldown (days)

Typical Fuel Transient Temperatures during DLOFC (MHTGR)

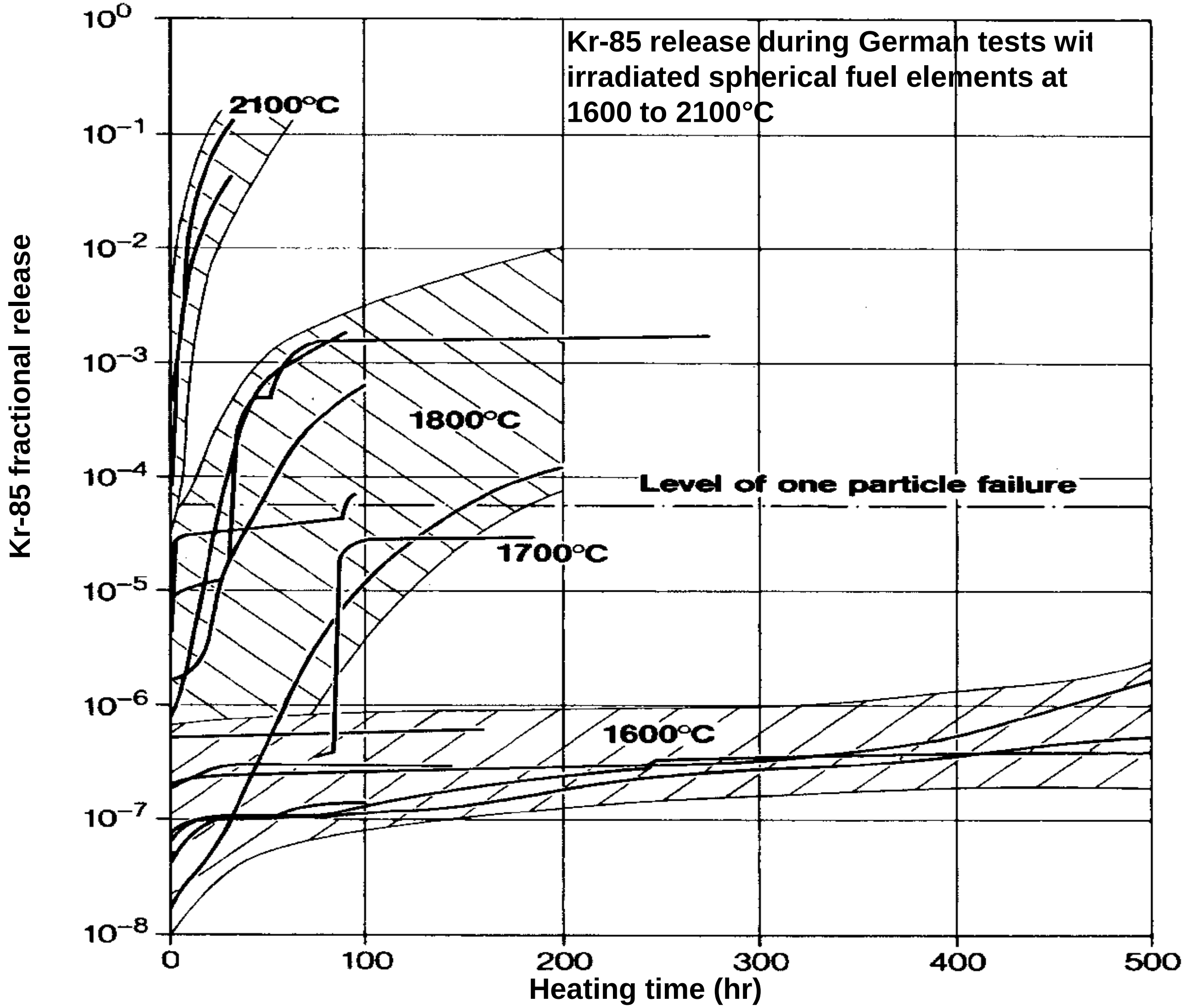


Acceptable Peak Reactor Core Temperatures at Worst Axial Location Several Days after Depressurized Loss of Forced Cooling



RSR: Removable Side Reflector
PSR: Permanent Side Reflector

Fuel Particles Are Highly Retentive at 100s of Degrees Above Normal Operation



Control of Heat Generation Accomplished by Intrinsic Shutdown and Reliable Control Material Insertion

- Large negative temperature coefficient intrinsically shuts reactor down
- Two independent and diverse systems of reactivity control for reactor shutdown drop by gravity on loss of power
  - Control rods
  - Reserve shutdown system
- Each system capable of maintaining subcriticality
- One system capable of maintaining cold shutdown during refueling
- Relied on for spectrum of off-normal events, such as rod withdrawal or water ingress

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Control of Air Ingress Assured by Inherent Characteristics and Passive Design Features

- Non-reacting helium coolant
- High integrity nuclear-grade pressure vessels make large break exceedingly unlikely
- Slow oxidation rate (high purity nuclear-grade graphite)
- Limited by core flow area and friction losses
- Graphite fuel element, embedded fuel compact matrix, and ceramic coatings protect fuel particles
- Reactor building embedment and vents that close after venting limit potential for gas mixture air in-leakage

NGNP Next Generation Nuclear Plant

Control of Moisture Ingress Assured by Inherent Characteristics and Design Features

- Non-reacting helium coolant
- Limited sources of water:
  - Moisture monitors
  - Steam generator isolation (does not require AC power)
  - Steam generator dump system
- Water-graphite reaction:
  - Endothermic
  - Slow reaction rate
- Graphite fuel element, fuel compact matrix, and ceramic coatings protect fuel particles

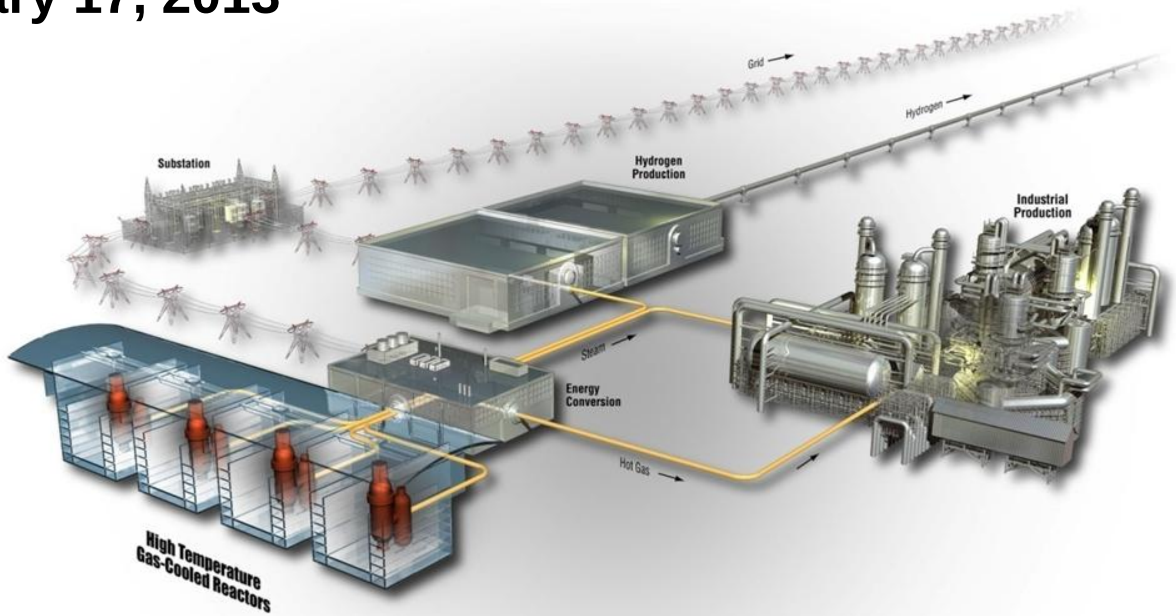
Safety Design Approach Summary

- Top objective is to meet the EPA PAGs at the EAB for spectrum of events within and beyond the design basis
- Responsive to Advanced Reactor Policy
- Modular HTGR designs employ multiple barriers to meet radionuclide retention requirements
  - Fuel Elements
    - Fuel kernels
    - Particle coatings (most important barrier)
    - Compact matrix and fuel element graphite
  - Helium coolant pressure boundary
  - Reactor building
- Retention of radionuclides at the source within ceramic fuel
  - Passive heat removal
  - Control of heat generation
  - Control of chemical attack

Licensing Basis Event Selection Process

ACRS Future Plant Designs Subcommittee Meeting

January 17, 2013



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Meeting Agenda

- HTGR Safety Design Bases
- Licensing Basis Event (LBE) Selection Process
- Functional Containment Performance and Mechanistic Source Terms
- Siting Source Terms
- Fuel Qualification and Radionuclide Retention

Outline – LBE Selection Process

- Risk-Informed, Performance-Based (RIPB) Framework and Top Level Regulatory Criteria (TLRC)
- LBE Categories and Frequency-Consequence (F-C) Curve
- Modular High-Temperature Gas-Cooled Reactor (MHTGR) Event Examples
- LBE Evaluation Structure
- Structures, Systems, and Components (SSC) Safety Classification

Requested Staff Positions – RIPB Topics

- Agree with the placement of top level regulatory criteria (TLRC) on a frequency-consequence (F-C) curve
- Establish frequency ranges based on mean event sequence frequency for the LBE event categories
- Endorse the “per plant-year” method for addressing risk at multi-reactor module plant sites
- Agree on key terminology and naming conventions for event categories
- Agree on the frequency cutoffs for the Design Basis Event (DBE) and Beyond Design Basis Event (BDBE) regions
- Endorse the overall process for performing assessments against TLRC, including issues with uncertainties and the probabilistic risk assessment (PRA), the calculational methodologies to be employed (conservative vs. best estimate), and the adequate incorporation of deterministic elements
- Endorse the proposed process and categorizations for structures, systems, and components (SSC) classification

Next Generation Nuclear Plant

Why Define Event Sequences Through LBE Selection Process?

- Technology neutral
- Comprehensive method for plant design and licensing to assure protection of the public for a spectrum of events
- Single failure criteria and associated redundancy may mask risk-significant accident sequences with multiple failures
- Quantitative; safety margins can be assessed

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RIPB Framework

- **What** must be met:
  - Top Level Regulatory Criteria (TLRC)
- **When** TLRC must be met:
  - Licensing Basis Events
- **How** TLRC must be met:
  - Safety Functions
  - SSC Safety Classification
- **How well** TLRC must be met:
  - Deterministic Design Basis Accidents (DBAs)
  - Defense-in-Depth
  - Regulatory Special Treatment

Next Generation Nuclear Plant

Bases for Top Level Regulatory Criteria (TLRC)

- Generic, technology-neutral and independent of plant site
- Quantitative
- Direct statements of acceptable consequences or risks to the public or the worker

TLRC for Protection of the Public

- 10CFR20 annualized offsite dose guidelines
  - 100 mrem/yr total effective dose equivalent
  - Measured on a cumulative basis annually at the EAB of the site
  - For normal operation and anticipated operational occurrences
- 10CFR50.34 (10CFR52.79) accident offsite doses
  - 25 rem total effective dose equivalent
  - Evaluated at the site EAB at 2 hour and at the site LPZ at 30 days
  - Design basis for off-normal events
- EPA Protective Action Guides (PAGs) offsite doses
  - 1 rem total effective dose equivalent for sheltering
  - Evaluated at the site EPZs
  - Emergency planning and protection during off-normal events
- NRC Safety Goals individual fatality risks
  - Prompt Quantitative Health Objective (QHO) of 5×10^{-7}/yr latent QHO of 2×10^{-6}/yr Evaluated at 1 mile for prompt and 10 miles for latent
  - Overall assurance of negligible cumulative risks during normal operation and off-normal events

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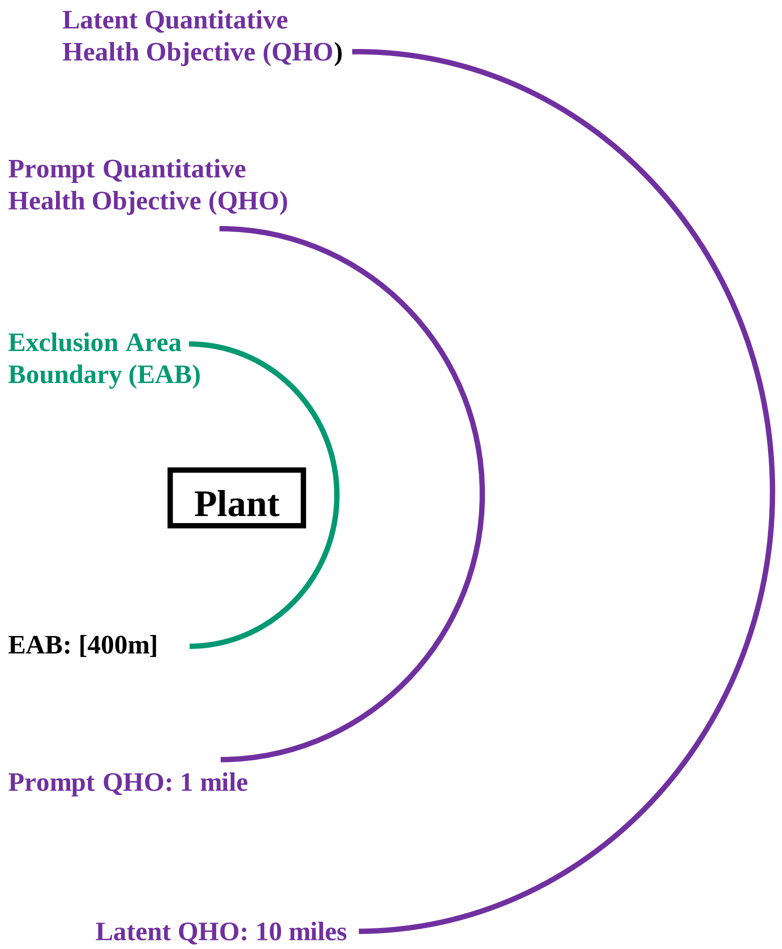
Selection of Frequency Axis for TLRC Placement

- Use of a risk assessment process leads to a frequency-consequence (F-C) curve construct
- Event likelihood is implicit in the current regulations; however, event frequencies are not typically stated
- **Event sequence frequency** is used since it is the frequency to be compared to the doses of the TLRC and the frequency for the NRC safety goal QHOs that are expressed as risks
- **Mean frequency** is selected as the best single measure of the expected outcome
- Event frequencies are expressed on a **per plant year basis**:
  - This is the important measure to the public (not whether a radionuclide release originated from one particular reactor module or system)
  - Provides the flexibility for the consequence limits in the TLRC to be met for **one or more reactor modules**

NGNP Next Generation Nuclear Plant

Selection of Consequence Axis for TLRC Placement

- Mean TEDE dose selected for consequence measure
- The Exclusion Area Boundary (EAB) was selected based on the following considerations:
  - It is the distance specified for the 10CFR20 and one of the 10CFR50.34 dose limits
  - Design objective is to meet the PAGs at the EAB to avoid public sheltering during off-normal events: the goal is for LPZ and EPZs to be at the same distance as the EAB (approximately 400m)
  - If met, the plant will have large margins to the average individual risk QHOs as measured within annular regions from the EAB to 1 and 10 miles, respectively
  - Supports co-location with industrial facilities



Event Selection Timeline

LBE evolution by design phase:

Deterministic LBEs:
PCC and DCC, etc.

Revise LBEs:
New initiating events, sequences, families, frequencies revised, etc.

Updated LBEs:
frequencies, sequences, etc.

Confirm LBEs:
confirm frequencies, sequences, etc.

Design Phases:
Preconceptual | **Conceptual** | **Preliminary** | **Final**

LBE selection process inputs vary by design phase:

Preconceptual:
- Initial design concept*
- Prior HTGR experience*
- Expert insights*

Conceptual:
- Basic design*
- Initial analyses (FMEA, scoping PRA, etc.)*
- Prior HTGR experience*
- Design rqmts.*
- Expert reviews*

Preliminary:
- Updated design*
- Detailed FMEAs, etc.*
- Initial PRA results*
- Expert reviews*
- Regulator interaction*

Final:
- Mature design
- Detailed FMEAs, etc.
- Complete PRA results
- Expert reviews
- Regulator feedback

* **Steps actually performed during MHTGR project through early preliminary design**

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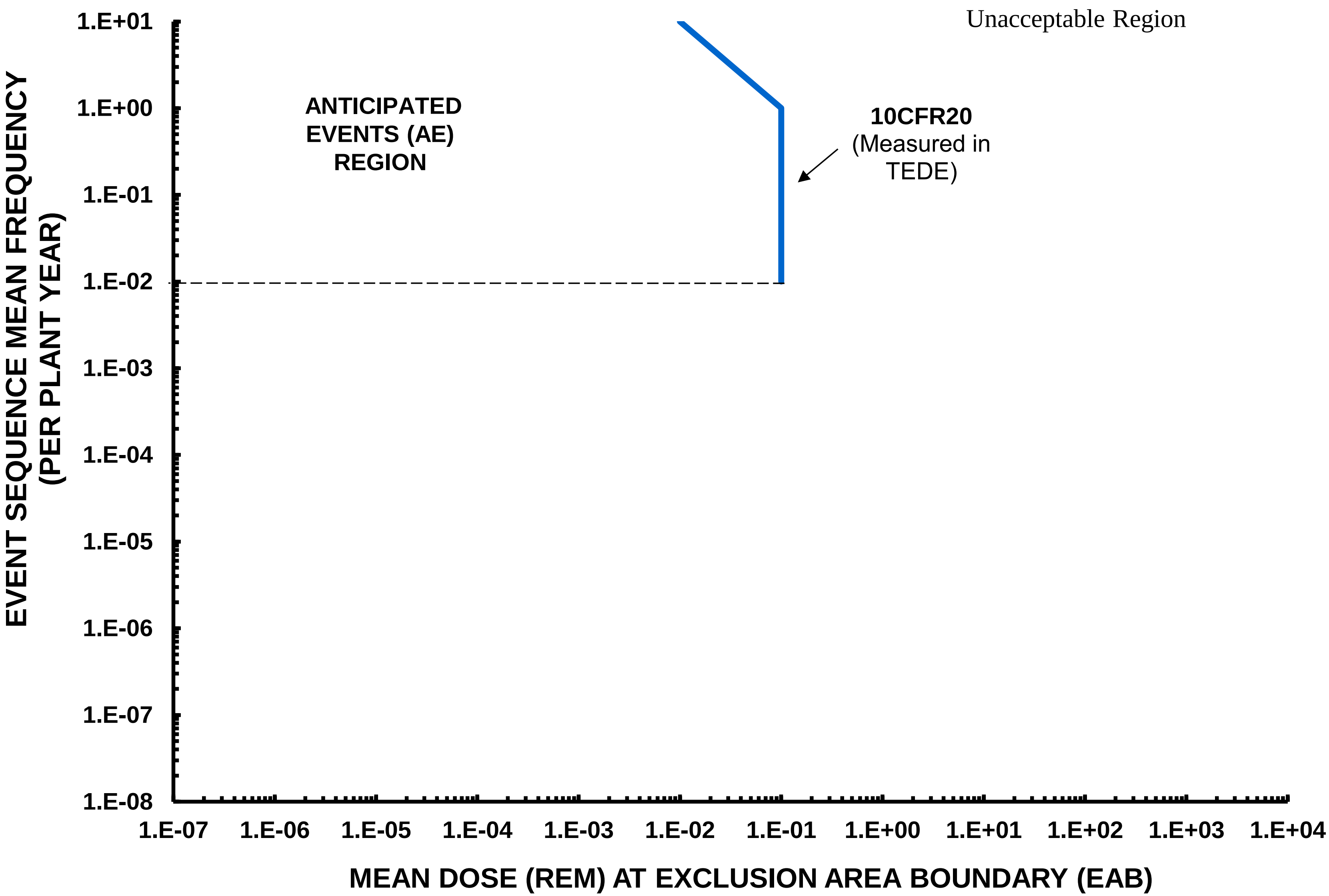
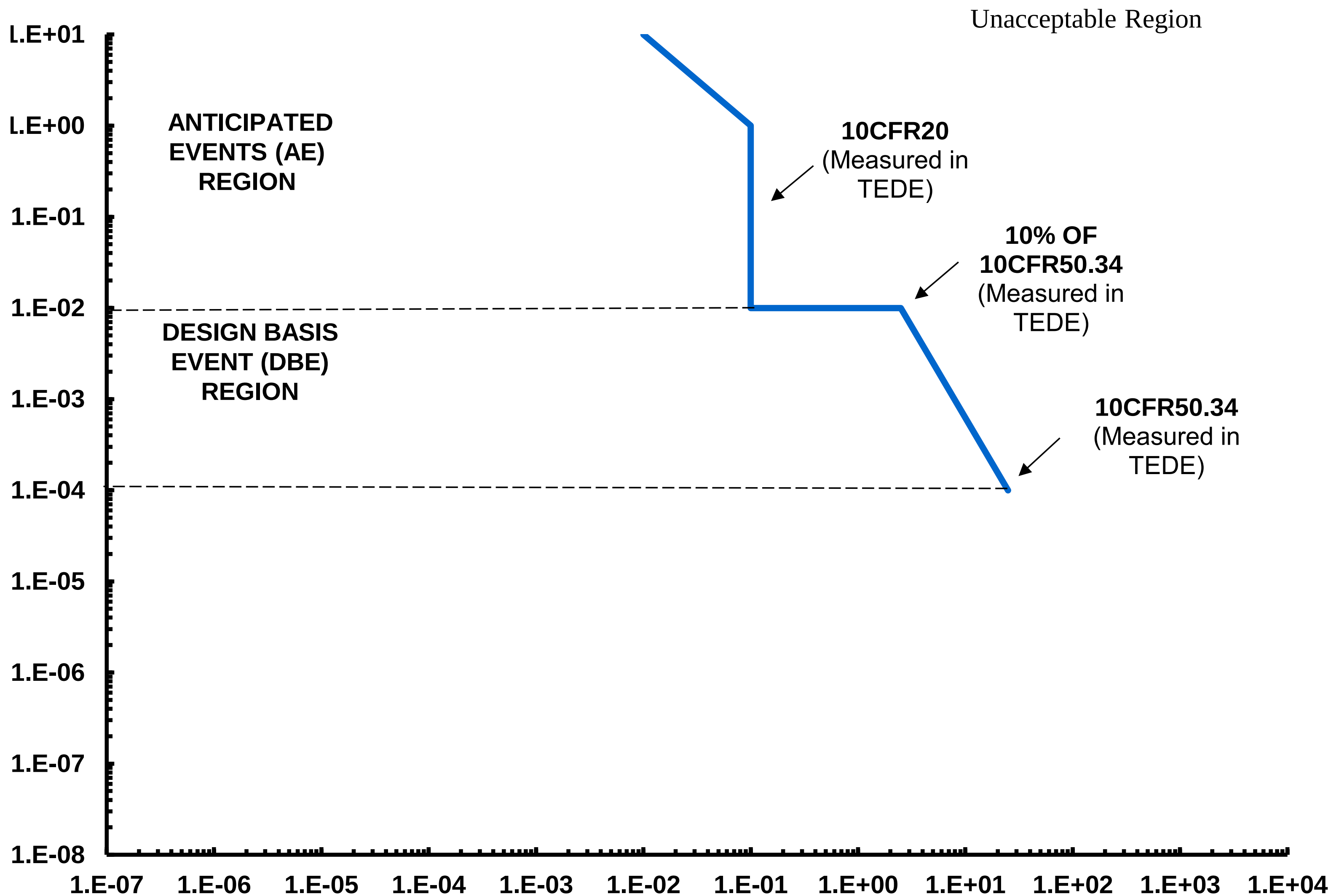
Categories of Licensing Basis Events

- TLRC apply to the full spectrum of normal operation and off-normal events
- Some specific TLRC apply to normal operation and anticipated events; others to design basis events; others to events less frequent than design basis events
- LBE categories selected:
  - Anticipated Events (AEs): >10^{-2}/plant year
  - Design Basis Events (DBEs): 10^{-2} to 10^{-4}/plant year
  - Beyond Design Basis Events (BDBEs): 10^{-4} to 5×10^{-7}/plant year
  - Design Basis Accidents (DBAs)
- Design Basis Accidents (analyzed in Chapter 15 of SARs) are deterministically derived from DBEs, assuming that only SSCs classified as safety-related are available
  - The event sequence frequency for some of these DBAs are expected to fall in or below the BDBE region

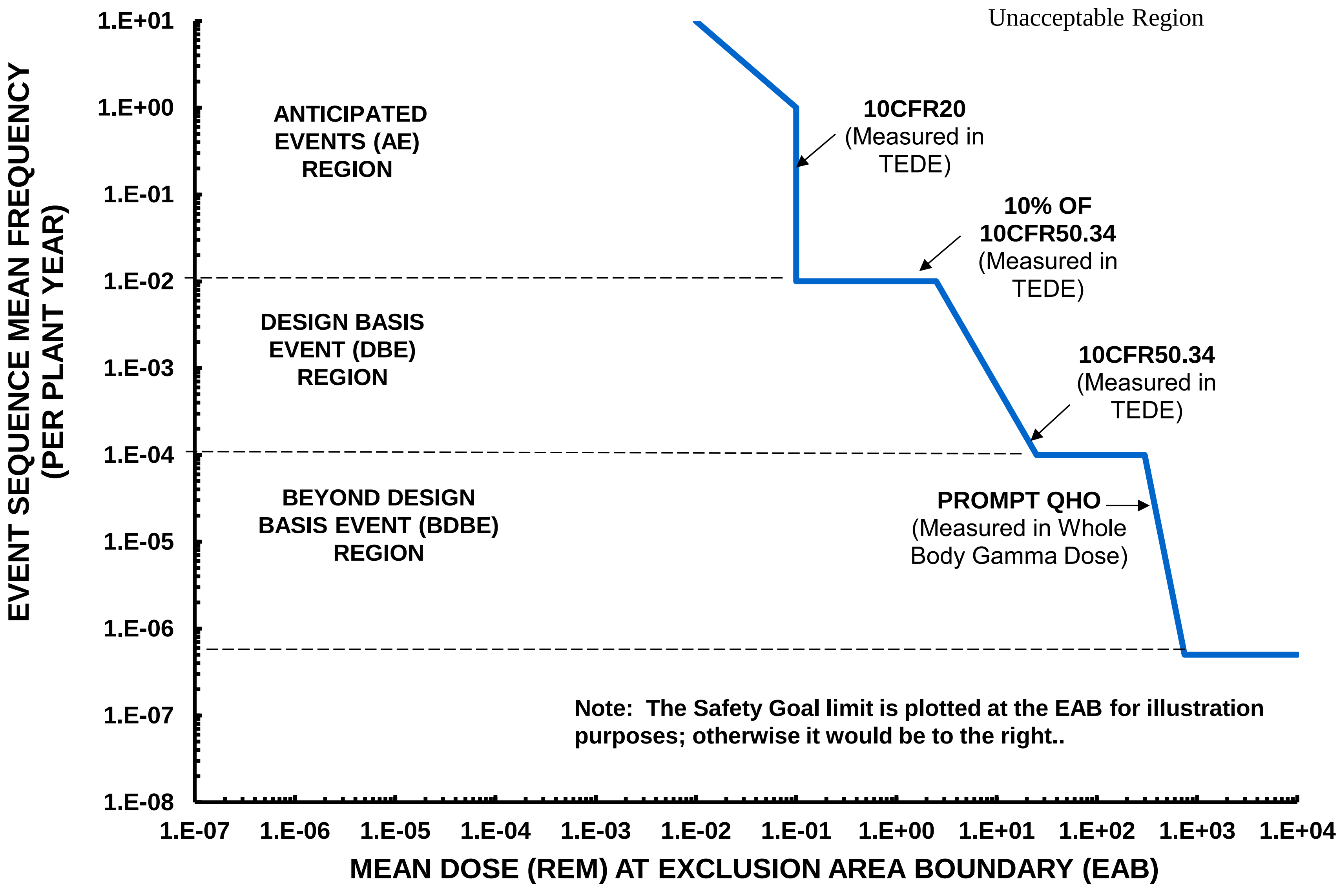
AE Region on F-C Curve



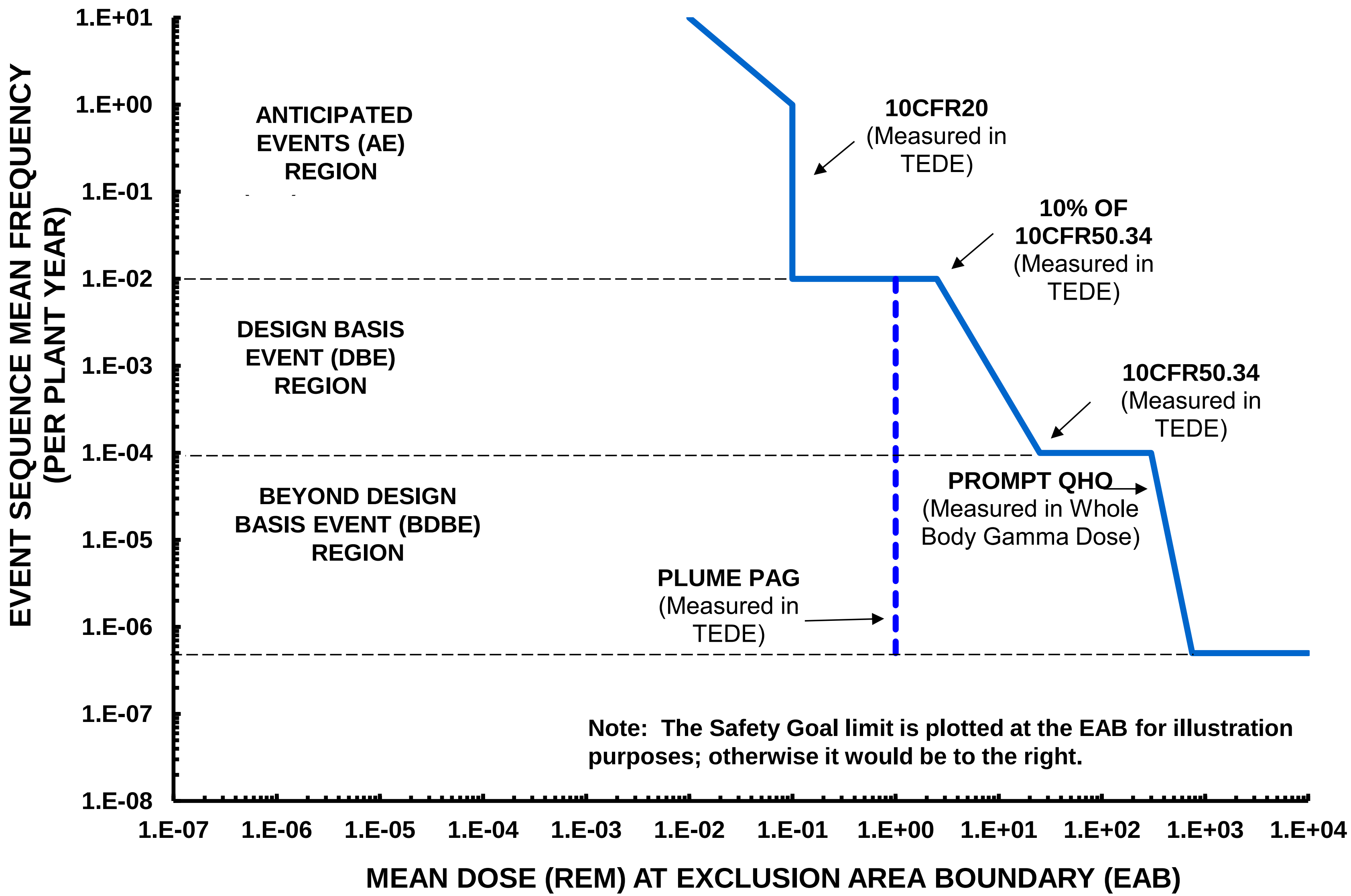
DBE Region on F-C Curve



BDBE Region on F-C Curve



NGNP F-C Curve



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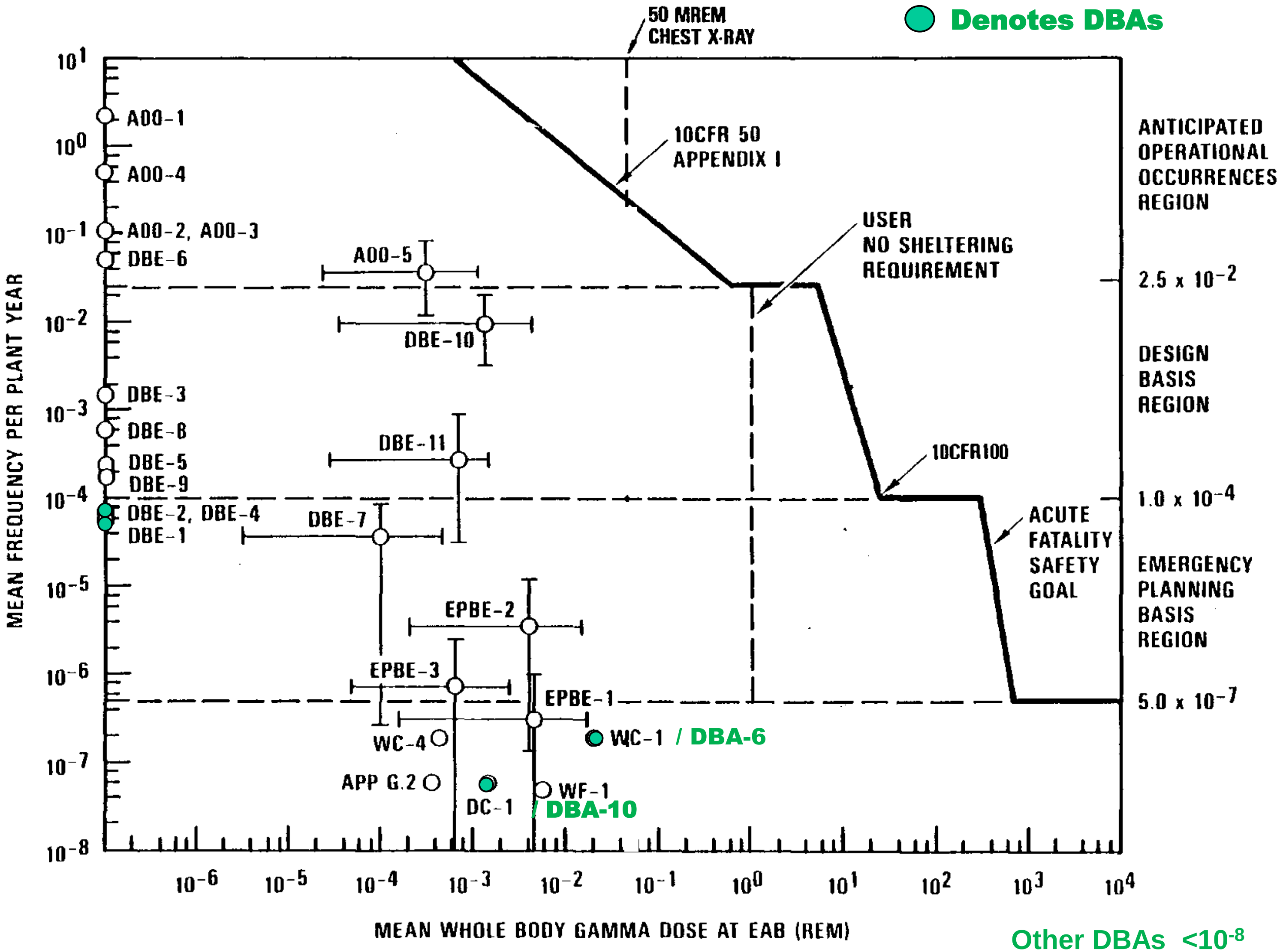
Limiting LBEs from MHTGR

- The MHTGR PSID identified several DBEs/DBAs and BDBEs enveloped by the following highest offsite consequence DBAs:
  - DBA-6: Steam Generator (SG) offset tube rupture with SG isolation and immediate and indefinite loss of forced cooling leading to an early (min to hr) and a delayed (days) radionuclide release from Helium Pressure Boundary (HPB) via opening of Vessel System (VS) relief valve to the Reactor Building (RB)
  - DBA-10: VS relief line breach of HPB with immediate and indefinite loss of forced cooling leading to an early (sec to min) and a delayed (days) radionuclide release from HPB to RB
  - DBA-11: Instrument line leak in HPB with immediate and indefinite loss of forced cooling leading to an early (min to hr) and a delayed (days) radionuclide release from HPB to RB

MHTGR DBEs, DBAs, and BDBEs (aka EPBEs) on F-C Plot *(circa 1987)*



Other DBAs $<10^{-8}$

Outline – LBE Selection Process

- Risk-Informed, Performance-Based (RIPB) Framework and Top Level Regulatory Criteria (TLRC)
- LBE Categories and Frequency-Consequence (F-C) Curve
- Modular High-Temperature Gas-Cooled Reactor (MHTGR) Event Examples
- LBE Evaluation Structure
- Structures, Systems, and Components (SSC) Safety Classification

LBE Evaluation Structure

Event Category/Type	10CFR20 – 0.1 rem	10CFR50.34 – 25 rem	EP PAGs – 1 rem	QHOs – Individual Risks
AEs	Mean Cumulative @ EAB			Mean Cumulative @ 1 and 10 miles
DBEs		Upper Bound @ EAB	Mean @ EPZ*	Mean Cumulative @ 1 and 10 miles
BDBEs			Mean @ EPZ*	Mean Cumulative @ 1 and 10 miles
DBAs		Upper Bound @ EAB		

***Design Objective: EPZ = EAB**

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Treatment of Uncertainties

- The mean and upper bound consequences are explicitly compared to the consequence criteria in all applicable LBE regions
- Example of parameters considered in the treatment of uncertainty applicable to HTGR consequence analysis:
  - Fuel inventory, circulating inventory, and plateout inventory
  - Initial fraction of defective fuel particles
  - Releases from defective fuel particles
  - Reactor building deposition and leakage
- The consequence uncertainty model accounts for the release and transport of radionuclides to the atmosphere from:
  - Fuel particle kernel
  - Silicon carbide and pyrocarbon coatings of the fuel particle
  - Fuel matrix and fuel element graphite
  - Helium pressure boundary
  - Reactor building

Key Features of Modular HTGR PRA

- All sources of radioactive material addressed
- Success criteria reflect reactor's unique features:
  - Reactor specific criteria to establish safe, stable end states
  - Breaches in HPB do not result in loss of cooling
  - Need functional basis for long system mission times
  - Plant response to ATWS and SBO fundamentally different than for LWR
- Smaller number of systems to model
- Integrated event sequence model for treatment of internal and external events and all operating and shutdown modes
- Source term phenomena unique to HTGRs
- Absence of severe core damage LWR-specific phenomena
- No "core damage" or "large early release" pinch points; CDF and LERF not applicable
- Unique HTGR end states covering a range of radionuclide release categories
- Address integrated risk of multi-reactor module plant
- Address sequences to support application and ensure no cliff edge effects
- Address hazards from nearby industrial facilities

Outline – LBE Selection Process

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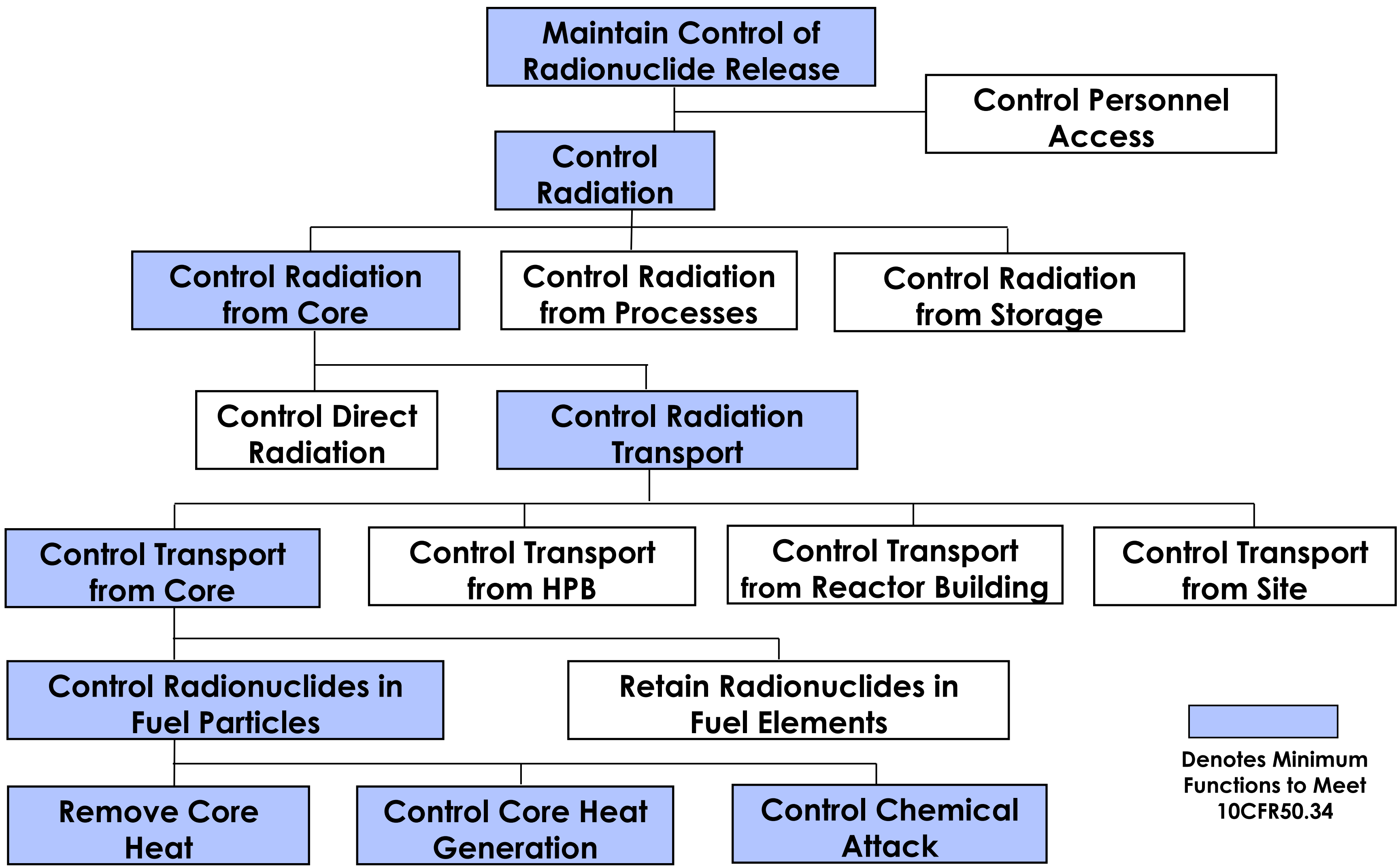
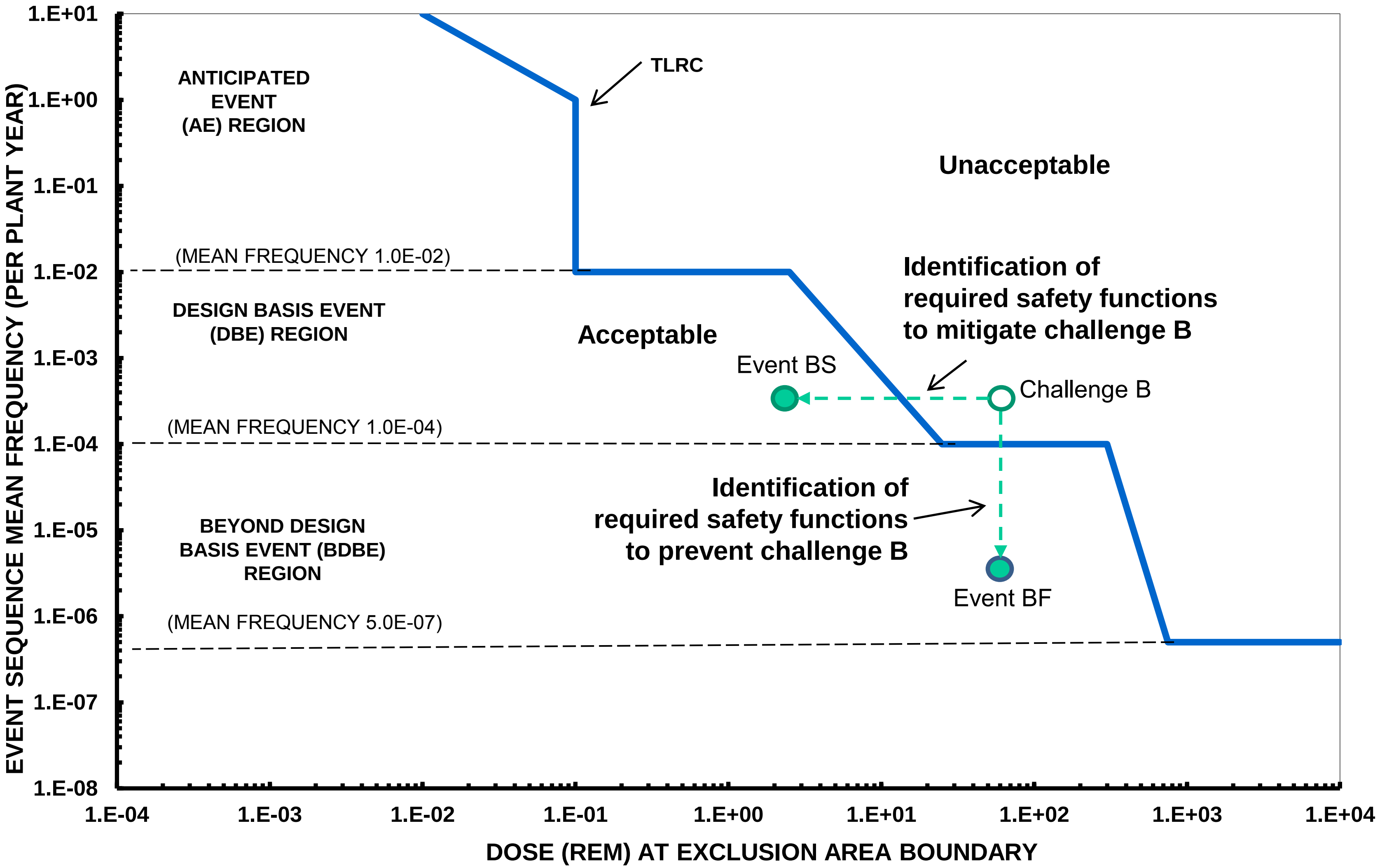
Process for SSC Classification as Safety-Related

- Determine the required safety functions for DBEs and BDBEs
- For each required safety function, determine which SSCs are available and have sufficient capability and reliability to meet the required safety function
- From this review, classify a set of SSCs as safety-related to assure that the required safety functions are accomplished

Functions for Control of Radionuclide Release



Identification of Safety Functions Leading to Safety-Related SSCs



MHTGR Example of Safety Classification for Core Heat Removal Function (1/2)

Are SSCs Available and Sufficient to Remove Core Heat in the DBE?	
Alternative Sets of SSCs	**DBE 11**
Initiating Event	**HPB small leak**
Reactor HTS ECA	**No**
Reactor SCS SCWS	**No**
Reactor RV RCCS	**Yes**
Reactor RV RB	**Yes**

Next Generation Nuclear Plant

MHTGR Example of
Safety Classification for Core Heat Removal Function (2/2)

Are SSCs Available and Sufficient to Remove Core Heat in the DBE?										
Alt. Sets of SSCs	**DBE 1**	**DBE 2**	**DBE 3**	**DBE 4**	**DBE 5**	**DBE 6/7**	**DBE 8/9**	**DBE 10**	**DBE 11**	**SSCs Classified as SR?**
IE	Transt, (LOSP +TT)	ATWS	Control rod withdwl	Control rod withdwl	SSE	SG tube rupture	SG tube leak	HPB moderate leak	HPB small leak	
Reactor HTS ECA	No	No	No	No	No	No	No	No	No	
Reactor SCS SCWS	No	Yes	Yes	No	Yes	Yes	Yes	Yes	No	
Reactor RV RCCS	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes, optimum selection to achieve capability and reliability**
Reactor RV RB	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	

LBE Selection Summary

- Licensing Basis Events determine ***when*** Top Level Regulatory Criteria must be met
- Selected during design and licensing process with risk insights from comprehensive full scope PRA that treats uncertainties
- Include AEs (expected in life of plant), DBEs (not expected in plant lifetime), BDBEs (not expected in fleet of plant lifetimes) and DBAs (Ch 15 events derived from DBEs with only safety related SSCs available)
- Safety classification determined by examining SSCs available and sufficient to successfully perform required safety functions to mitigate spectrum of DBEs

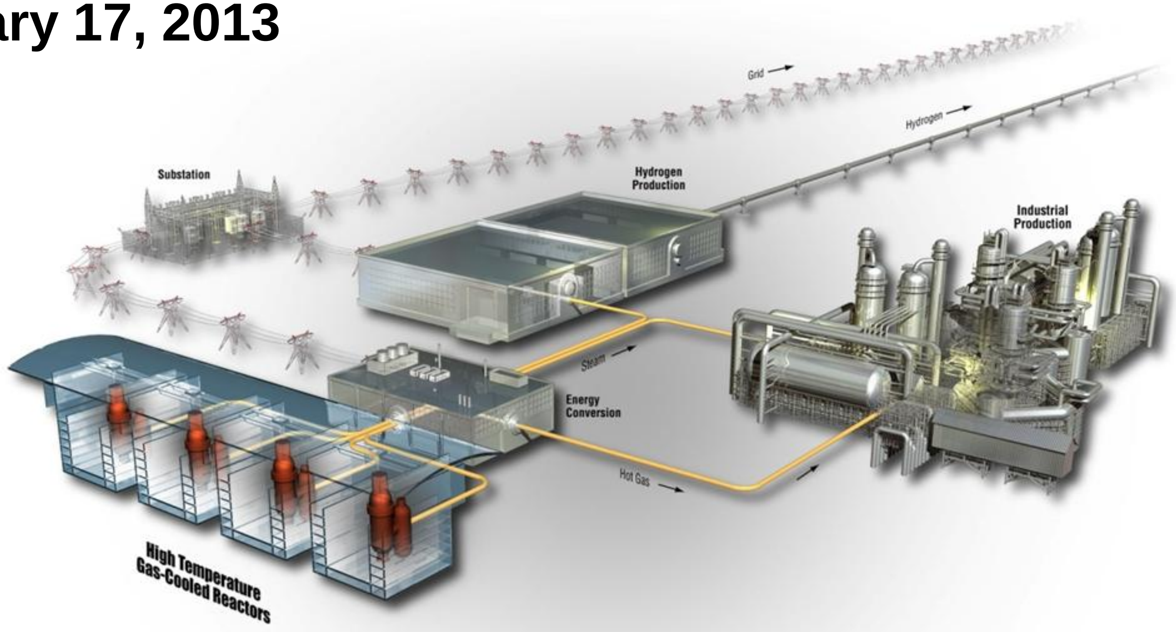
Requested Staff Positions – RIPB Topics

- Agree with the placement of top level regulatory criteria (TLRC) on a frequency-consequence (F-C) curve
- Establish frequency ranges based on mean event sequence frequency for the LBE event categories
- Endorse the “per plant-year” method for addressing risk at multi-reactor module plant sites
- Agree on key terminology and naming conventions for event categories
- Agree on the frequency cutoffs for the Design Basis Event (DBE) and Beyond Design Basis Event (BDBE) regions
- Endorse the overall process for performing assessments against TLRC, including issues with uncertainties and the probabilistic risk assessment (PRA), the calculational methodologies to be employed (conservative vs. best estimate), and the adequate incorporation of deterministic elements
- Endorse the proposed process and categorizations for structures, systems, and components (SSC) classification

Functional Containment Performance and Mechanistic Source Terms

ACRS Future Plant Designs Subcommittee Meeting

January 17, 2013



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INL Idaho National Laboratory

Meeting Agenda

- HTGR Safety Design Bases
- Licensing Basis Event (LBE) Selection Process
- Functional Containment Performance and Mechanistic Source Terms
- Siting Source Terms
- Fuel Qualification and Radionuclide Retention

Presentation Outline

- Introduction
- Regulatory Background
- Functional Containment Performance and Mechanistic Source Term Determination
- Conclusions

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INL/EXT-10-17997

Mechanistic Source Terms (MST) White Paper

July 2010

NRC ADAMS Accession Number: ML102040260


INL/EXT-10-17997
Revision 0

Mechanistic Source Terms White Paper

July 2010

The INL is a U.S. Department of Energy National Laboratory operated by Battelle Energy Alliance

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MST White Paper Contains Information on Radionuclide Transport and Retention in the Modular HTGR

- Functional containment description
- Radionuclide behavior in the fuel, primary circuit, and reactor building
- MST models and modeling assumptions
- Sources of data on radionuclide behavior
- Experimental methods for data collection

Requested NRC Staff Positions on Functional Containment (July 6, 2012 Letter)

- Item 1.b. Establish options regarding functional containment performance standards
  - as requested by the Commission in the Staff Requirements Memorandum (SRM) to SECY-03-0047, "Policy Issues Related to Licensing Non-Light Water Reactor Designs,”
  - and discussed further in SECY-05-0006, "Second Status Paper on the Staffs Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing"

Next Generation Nuclear Plant

Requested NRC Staff Positions on MST (July 6, 2012 Letter)

- Item 3.a: Endorse the proposed NGNP mechanistic source terms definition – the quantities of radionuclides released from the reactor building to the environment during the spectrum of LBEs, including timing, physical and chemical forms, and thermal energy of the release
- Item 3.b: Agree that NGNP source terms are event specific and determined mechanistically using models of radionuclide generation and transport that account for fuel and reactor design characteristics, passive features, and the radionuclide release barriers
- Item 3.c: Agree that NGNP has adequately identified the key HTGR fission product transport phenomena and has established acceptable plans for evaluating and characterizing those phenomena and associated uncertainties

Next Generation Nuclear Plant

- Introduction
- Regulatory Background
- Functional Containment Performance and Mechanistic Source Term Determination
- Conclusions

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Regulatory Precedent for Functional Containment and Mechanistic Source Terms

- Advanced Reactor Policy Statement
- SECY Documents
  - 93-092 05-0006
  - 95-299 10-0034
  - 03-0047
  - Other SECYs discuss “new,” “revised,” or “physically based” source terms for evolutionary and advanced LWRs
- US HTGR Licensing Interactions
  - Peach Bottom (Unit 1)
  - Fort St. Vrain
  - Large HTGR (to Construction Permit stage)
  - DOE MHTGR pre-application
  - Pebble Bed Modular Reactor pre-application submittals

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DOE MHTGR PSER NUREG 1338 Drafts: 1989 and 1995

- 1989 – (p 15-23) Section 15.6 “The staff has judged that the siting source term can be based on a mechanistic analysis of fuel failure and radionuclide inventory contained in the circulating helium or plated out within the primary system

  Final acceptance of a mechanistically calculated source term is dependent on satisfactory accomplishment of research and development goals, satisfactory resolution of the safety issues and deferred items, and a prototype test program demonstrating that the combination of research and development findings and analytical predictions confirm the staff's detailed and overall safety conclusions for the MHTGR”
- 1995 – (p 3-16) “Commission decided that a mechanistic source specific to the design was acceptable”

DOE MHTGR PSER NUREG 1338 Draft: 1995

- 1995 – (p 4-8) "In its decision on source terms for the advanced reactors policy issues…the Commission approved the use of mechanistic source terms for the MHTGR"
- "However, the Commission criteria for use of mechanistic source terms is that the source terms had to be based on:
  - The fuel performance being well understood,
  - Fission-product transport being adequately modeled, and
  - Events considered in the development of source terms include bounding severe accidents and design-dependent uncertainties"

DOE MHTGR PSER NUREG 1338 Draft: 1995, cont'd

- 1995 – (p 4-11) "…the Commission decided that a conventional LWR, leaktight containment should not be required for advanced reactor designs. It approved the use of containment functional design criteria for evaluating the acceptability of proposed containment designs rather than the use of prescriptive design criteria"
- 1995 – (p 5-10) "[The] position regarding containment allows the acceptance of containments with leak rates that are not 'essentially leaktight' as described in GDC 16 for LWRs"

- Introduction
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- Functional Containment Performance and Mechanistic Source Term Determination
- Conclusions

What is the “Functional Containment?”

The collection of design selections that, taken together, ensure that:

- Radionuclides are retained within multiple barriers, with emphasis on retention at their source in the fuel, and
- Regulatory requirements and plant design goals for release of radionuclides are met at the Exclusion Area Boundary

HTGRs have Multiple Barriers to Radionuclide Release that Comprise the "Functional Containment"

- Fuel Kernel
- Fuel Particle Coatings
- Matrix/Graphite

Fuel Element

- Helium Pressure Boundary

- Reactor Building

Modular HTGR Source Term Definition

- Quantities of radionuclides released from the reactor building to the environment during Licensing Basis Events. This includes timing, physical and chemical forms, and thermal energy of the release
- Modular HTGR Source Terms are:
  - Event-specific
  - Determined mechanistically using models of fission product generation and transport that account for reactor inherent and passive design features and the performance of the fission product release barriers that comprise the functional containment
  - Different from the LWR source term that is based on a severe core damage event

Modular HTGR Source Term Analysis

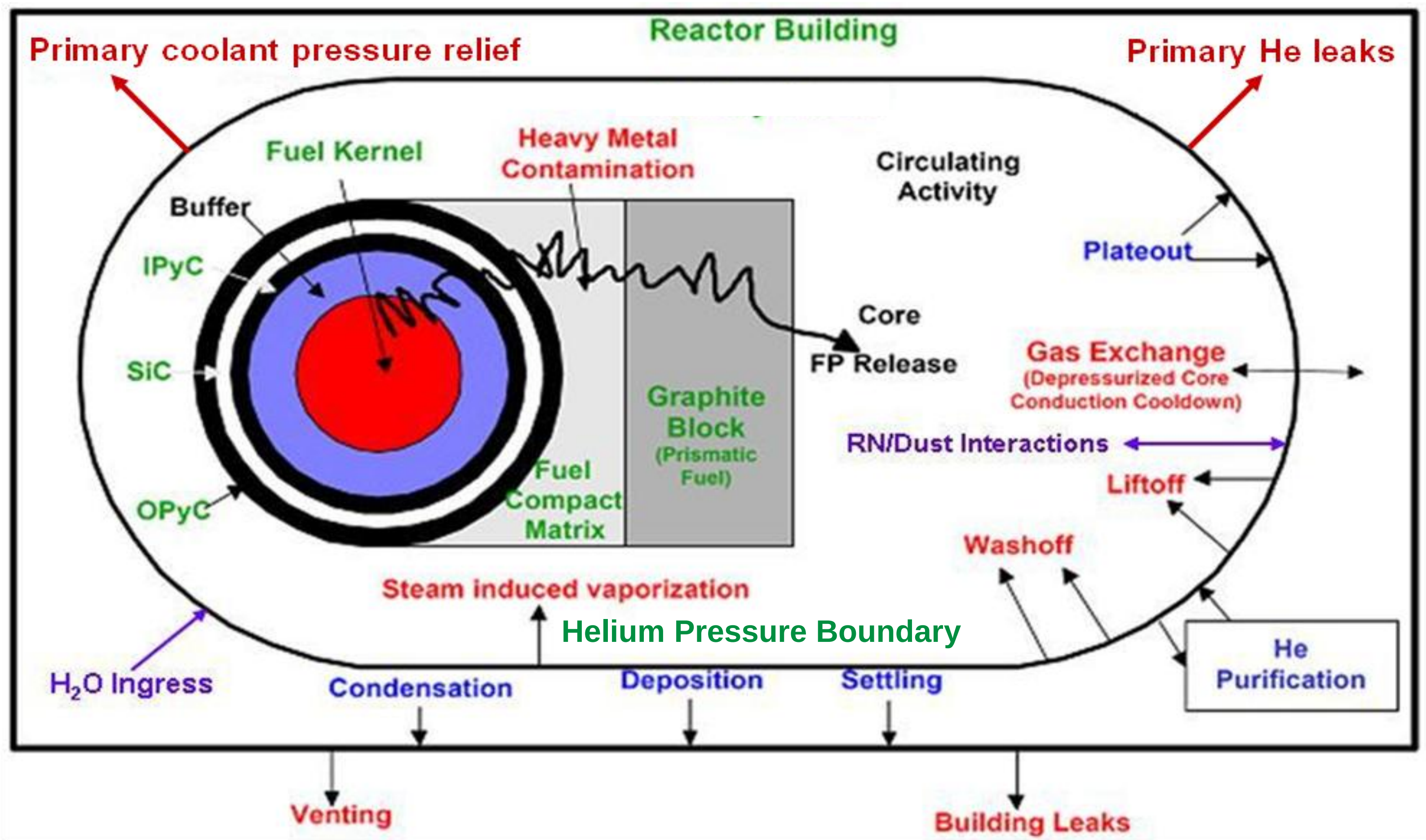
- Considers hundreds of radionuclides
- To facilitate analysis, fission products are grouped by chemical similarity and by similarity in transport properties
- Experience based on past analyses suggest that I-131, Cs-137, Cs-134 and Sr-90 are dominant contributors to offsite dose

Fission Product Transport Models Mechanistically Calculate

- Transport of radionuclides from their point of origin through the fuel to the circulating helium
- Circulating activity in the HPB
- Distribution of condensable radionuclides in the HPB
- Radionuclide release to and distribution in the reactor building
- Radionuclide release from the reactor building to the environment (source term)

In addition to providing source terms, these calculations provide radionuclide inventories throughout the facility that can be used for other purposes (shielding, worker dose, equipment EQ, etc.)

Modular HTGR Fission Product Retention



The phenomena illustrated in this figure are modeled to determine mechanistic source terms for normal and off-normal events

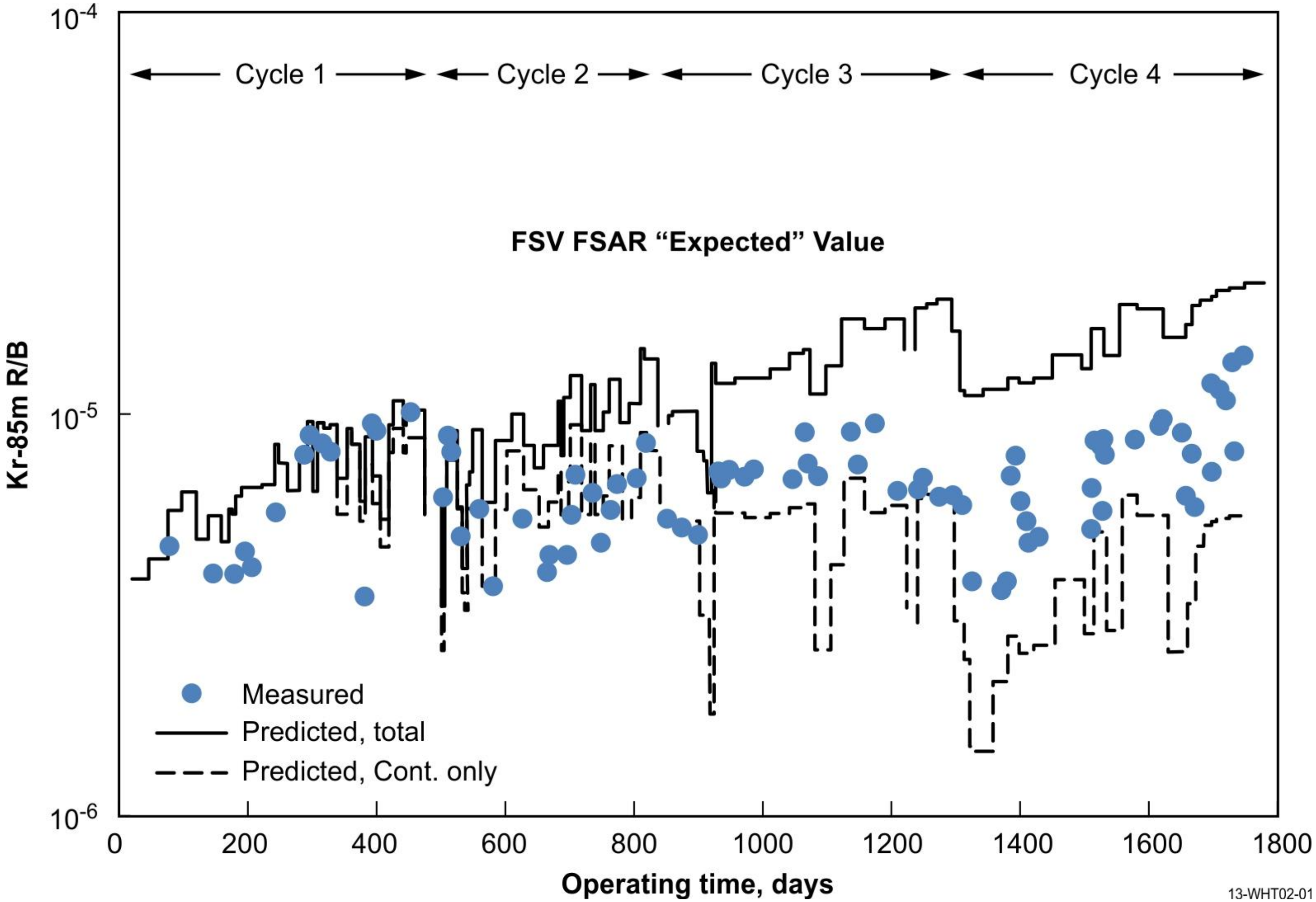
Fuel Particle Coatings are the Primary Barrier to Radionuclide Release During Normal Operation and Off-Normal Events

- Low heavy metal contamination and low initially defective fuel particles in as-manufactured fuel (~10^{-5})
- Minimal radionuclide release from incremental fuel failure during normal operation ($<10^{-4}$)
- Minimal radionuclide release from incremental fuel failure during Licensing Basis Events ($<10^{-4}$)
- Radionuclide release during LBEs dominated by exposed heavy metal (contamination and exposed fuel kernels)

Radionuclide Behavior During Normal Operation

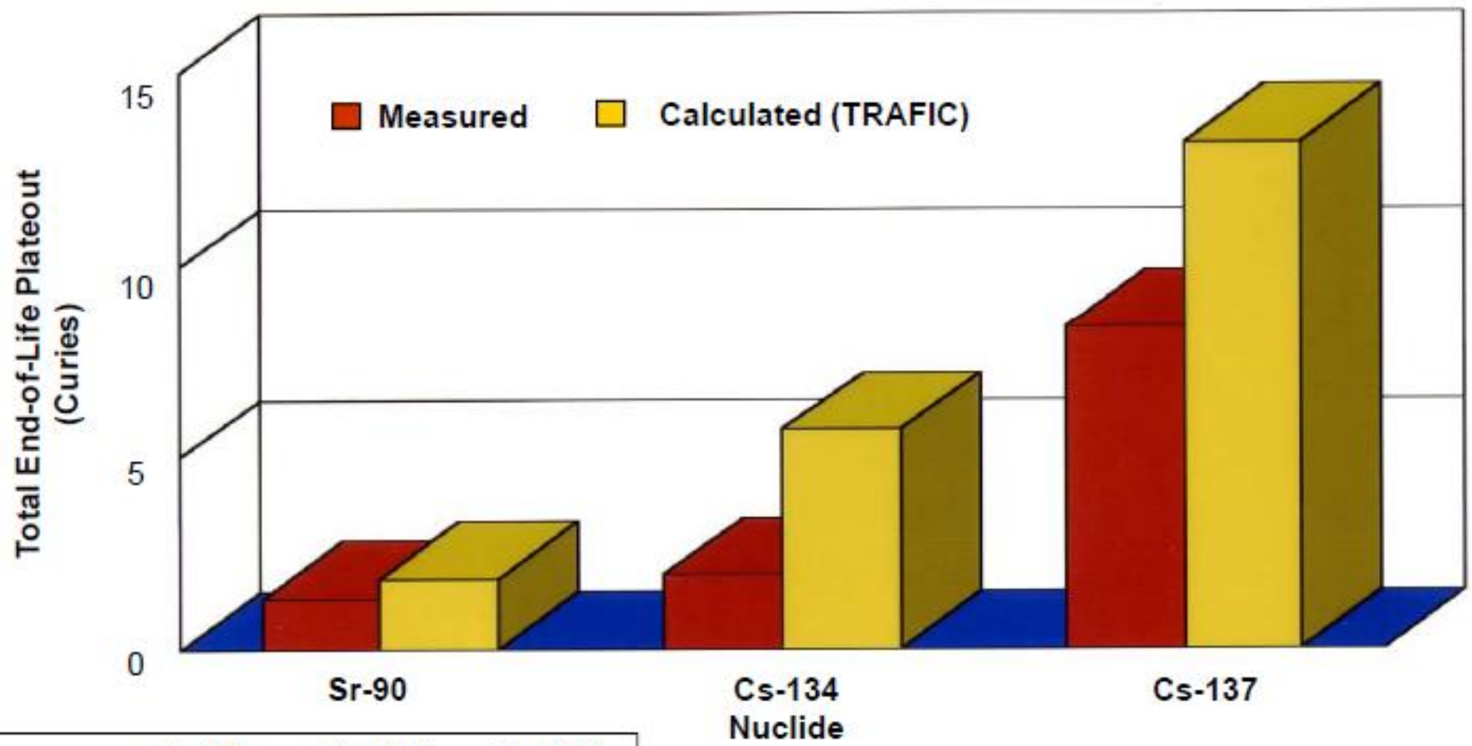
- Most radionuclides reach a steady state concentration and distribution in the primary circuit (long lived isotopes like Cs-137 and Sr-90 are exceptions – plateout inventory builds up over plant life)
- Concentration and distribution are affected by:
  - Radionuclide half-life
  - Initial fuel quality
  - Incremental fuel failure during normal operation
  - Fission product fractional release from fuel kernel
  - Transport of fission products through particle coatings, matrix, and graphite
  - Fission product sorptivity on fuel matrix and graphite materials
  - Fission product sorptivity on primary circuit surfaces (plateout)
  - Helium purification system performance

Comparisons of Calculated and Measured Fission Gas Release: Fort St. Vrain Kr-85m R/B – Normal Operation



Calculations using SURVEY code at General Atomics

Comparisons of Calculated and Measured FSV Metallic Fission Product Release - Normal Operation



	Sr-90	Cs-134	Cs-137
Measured	1.3	1.9	8.4
Calculated	1.8	5.7	13.2

Calculations using TRAFIC Code at General Atomics

Early Release Mechanisms: Off-Normal Events

- Circulating activity
  - Released from HPB with helium in minutes to days as a result of HPB leak/break
  - Amount of release depends on location and any operator actions to isolate and/or intentionally depressurize
- Liftoff of plateout
  - For large breaks, fractional radionuclide amounts released from HPB with helium relatively quickly (minutes)
  - Amount of release depends on HPB break size and location. Surface shear forces must exceed those for normal operation to obtain liftoff
- HPB relief valve behavior
  - Sufficient moisture ingress can result in lifting of pressure vessel relief valve
  - Washoff of fractional radionuclide amounts can occur – can exceed liftoff fractions
  - Relief valve may cycle open/closed or may fail open

Delayed Release Mechanisms: Off-Normal Events

- Partial release from contamination, defective particles, particles failed in service, and particles that fail during off-normal events – tens of hours to days
- Delayed release from fuel is typically larger than circulating activity and any liftoff/washoff
- Amount of delayed release from fuel depends on time at temperature, level of oxidants, and radionuclide volatility
- Amount of delayed release from HPB depends on location and size of leak/break and on timing relative to expansion/contraction of gas mixture within the HPB
  - Small leaks have greater releases from HPB
  - Pressure relief valve behavior (reseating) affects release
  - Releases cease when temperatures within the HPB decrease due to core cooldown

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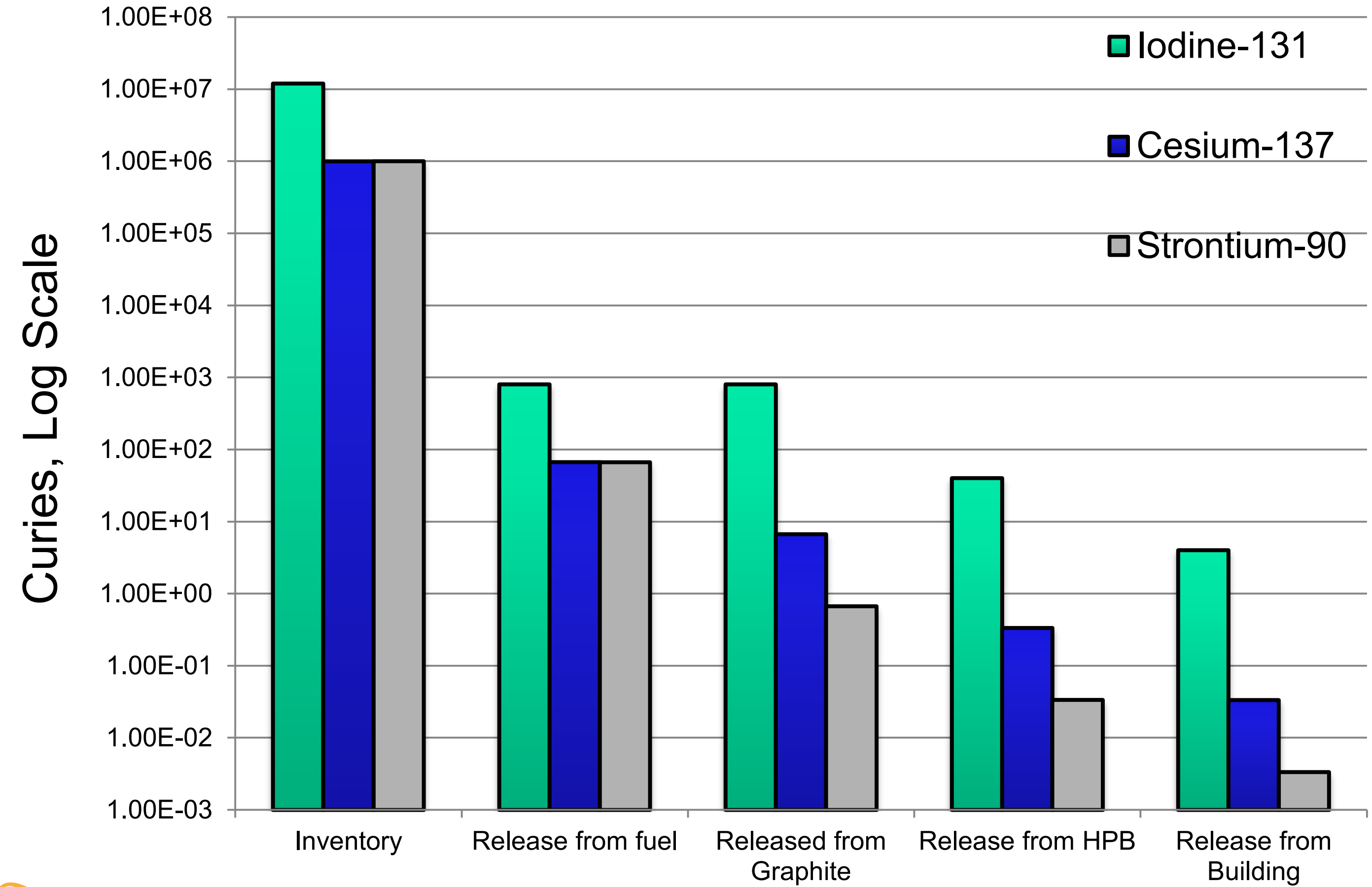
Representative Functional Containment Performance During a Depressurized Loss of Forced Cooling*



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*MHTGR PSID, 1989

Functional Containment Performance Summary

- Radionuclide retention within fuel during normal operation with relatively low inventory to HPB
- Limiting off-normal events characterized by
  - an initial release from the HPB depending on leak/break/pressure relief size
  - a larger, delayed release from the fuel
- Functional containment will meet 10CFR50.34 (10 CFR 52.79) at the EAB with margin for a wide spectrum of off-normal events without consideration of reactor building retention
- Functional containment (including reactor building) will meet EPA PAGs at the EAB with margin for wide spectrum of off-normal events

- Introduction
- Regulatory Background
- Functional Containment Performance and Mechanistic Source Term Determination
- Conclusions

The NGNP Approach to Functional Containment and Mechanistic Source Terms

- Is consistent with the NRC Advanced Reactor Policy Statement
- Is consistent with discussions of containment function and mechanistic source terms in various NRC SECY documents and with approaches previously reviewed by the NRC staff for modular HTGRs
- Is event specific and can be applied to the full range of licensing basis events
- Uses mechanistic models of fission product generation and transport that account for reactor inherent and passive design features and the performance of the fission product release barriers that comprise the functional containment

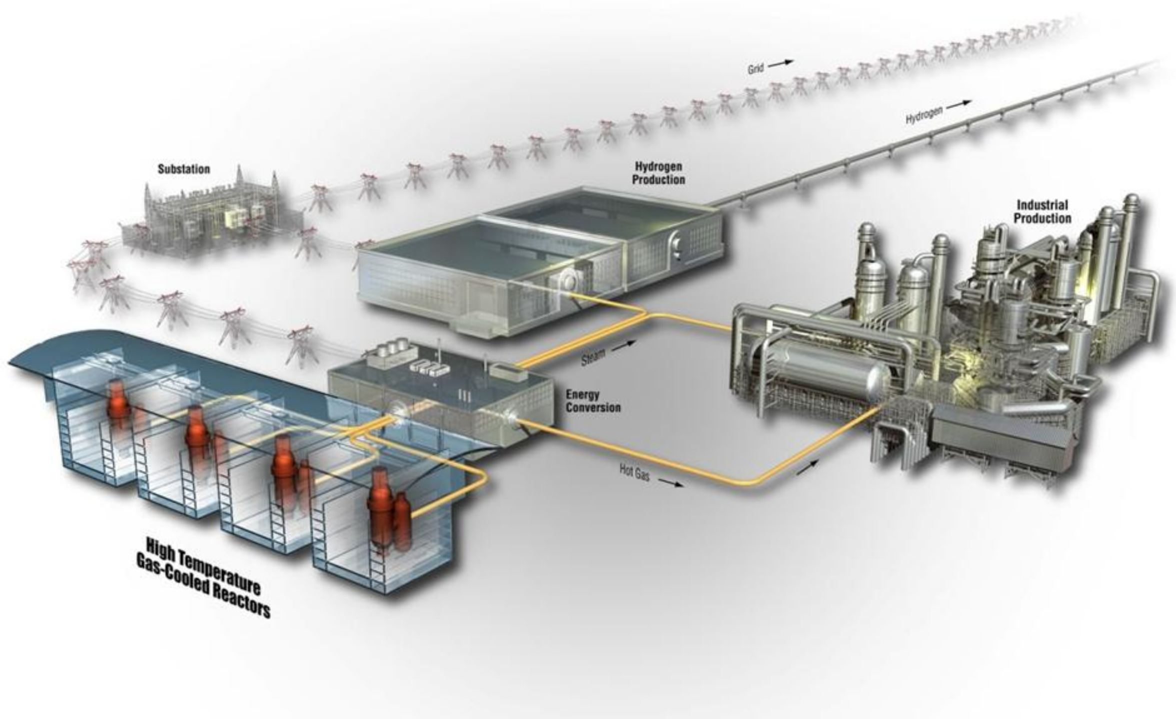
Requested NRC Staff Positions – Recap

- Item 1.b: Establish options regarding functional containment performance standards
- Item 3.a: Endorse the proposed NGNP mechanistic source terms definition
- Item 3.b: Agree that NGNP source terms are event specific and determined mechanistically
- Item 3.c: Agree that NGNP has adequately identified the key HTGR fission product transport phenomena and has established acceptable plans for evaluating and characterizing those phenomena and associated uncertainties

Siting Source Terms

ACRS Future Plants Design Subcommittee Meeting

January 17, 2013



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Meeting Agenda

- HTGR Safety Design Bases
- Licensing Basis Event (LBE) Selection Process
- Functional Containment Performance and Mechanistic Source Terms
- Siting Source Terms
- Fuel Qualification and Radionuclide Retention

Next Generation Nuclear Plant

Siting Source Terms Presentation Outline

- Requested NRC Staff Position Regarding Siting Source Terms
- NGNP Siting Source Terms Approach
- Event Sequences Involving Graphite Oxidation
- SST Conclusions

- Requested NRC Staff Position Regarding Siting Source Terms
- NGNP Siting Source Terms Approach
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- SST Conclusions

Requested NRC Staff Positions on Siting Source Terms (July 6, 2012 Letter)

- Item 1.c: Establish a staff position to support a final determination regarding how LBEs will be considered for the purpose of plant siting and functional containment design decisions, taking into consideration previous staff positions in SECY-95-299, that improved fuel performance is a justification for revising siting source terms and containment design requirements
  - In particular, we request that this staff position provide an adaptation of the guidance that has generally been applied to light water reactors (LWRs) for compliance with 10 CFR 100.21. (It is noted that for LWRs, this guidance has typically included the assumption of a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.)
  - The NRC's development of the NGNP adaptation of this guidance, which should reflect the NGNP's unique event response characteristics, will rely heavily on the establishment of the NRC staff positions associated with Licensing Basis Event Selection and establishing Mechanistic Source Terms

- Requested NRC Staff Position Regarding Siting Source Terms
- NGNP Siting Source Terms Approach
- Event Sequences Involving Graphite Oxidation
- SST Conclusions

NGNP Approach to SSTs

- NGNP's approach to SSTs is patterned after that developed by DOE and the NRC staff in the development and review in the late 1980s and early 1990s of the MHTGR Conceptual Design documents including the PSID and PRA
  - Develop the design consistent with the safety design approach
  - Utilize risk insights as input to the design for the range of user and regulatory requirements
  - Select and mechanistically evaluate risk-informed LBEs including DBEs/DBAs as well as BDBEs, against the Top Level Regulatory Criteria (10CFR20, 10CFR50.34 and 52.79, and Prompt QHO) and the NGNP design goal (PAG at EAB)
- Consistent with MST approach, mechanistically evaluate events over LBE-spectrum that have limiting dose consequences for use as SSTs

MHTGR DBA Examples

- The MHTGR PSID identified several DBEs/DBAs and BDBEs enveloped by the following highest offsite consequence DBAs:
  - DBA-6: Steam Generator (SG) offset tube rupture with SG isolation and immediate and indefinite loss of forced cooling leading to an early (min to hr) and a delayed (days) radionuclide release from Helium Pressure Boundary (HPB) via opening of Vessel System (VS) relief valve to the Reactor Building (RB)
  - DBA-10: VS relief line breach of HPB with immediate and indefinite loss of forced cooling leading to an early (sec to min) and a delayed (days) radionuclide release from HPB to RB
  - DBA-11: Instrument line leak in HPB with immediate and indefinite loss of forced cooling leading to an early (min to hr) and a delayed (days) radionuclide release from HPB to RB
- Each of these DBAs entails ingress of moisture or air into the reactor

Bounding Event Sequences will also be Considered for Cliff Edge Effects

- To assure that there are no cliff edge effects and to understand the safety capability of HTGRs, supplement the LBE-derived SSTs with insights from a best estimate mechanistic evaluation of bounding event sequences, with the understanding that:
  - Such events shall be physically plausible rather than non-physical, arbitrary combinations of event parameters or end-states
  - While the bounding event sequences would not be rigorously quantified in terms of frequency, <u>it is expected that they would generally have frequencies lower than the BDBE region</u>
  - Events and their evaluation will consider the intrinsic and passive characteristics and the safety behavior of the HTGR

Process for Selection of Bounding Event Sequences

- Bounding event sequences will be selected based on a deterministic review of physically plausible events that potentially impact HTGR safety functions:
  - Remove core heat
  - Control core heat generation
  - Control chemical attack (e.g., graphite oxidation)
- The initial selection of bounding event sequences requires completion of preliminary design
- The bounding event selection process will use as a starting point the six MHTGR bounding event sequences requested by NRC staff in MHTGR PSID RAIs

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MHTGR Bounding Event Sequences from NRC Staff

- BES-1 Inadvertent withdrawal of all control rods without scram for 36 hours (one module)
- BES-2 Station blackout (all modules) for 36 hours
- BES-3 Loss of forced cooling plus loss of RCCS for 36 hours (one module)
- BES-4 Steam generator tube rupture (25% of tubes) with failure to isolate or dump
- BES-5 Rapid depressurization (one module): double ended guillotine break of crossduct (sic) with failure to scram (assume RCCS failed for 36 hours and 25% unblocked thereafter)
- BES-6 External events consistent with those imposed on LWRs

Application of Bounding Event Sequence Analysis Results

- Analyses of bounding event sequences will be used to:
  - Identify and understand potential for “cliff edge effects” (i.e., high consequence events)
  - Determine potential risk significant plant or system vulnerabilities
  - Identify risk mitigation strategies as needed
- Analyses results will be documented as a part of the licensing application process

Previous NRC Staff Positions on MHTGR Bounding Event Sequences

- (1989) NUREG-1338
  - (p 15-7) – “The staff judges that these [bounding events proposed by the staff] results show that the MHTGR has the potential to cope with extremely rare and severe events without the release of a significant amount of fission products”
  - Appendix C, (p 4) – ACRS statement: “Neither the designers, the NRC staff, nor members of the ACRS have been able to postulate accident scenarios of reasonable credibility, for which an additional physical barrier to the release of fission products is required in order to provide adequate protection to the public”

- Requested NRC Staff Position Regarding Siting Source Terms
- NGNP Siting Source Terms Approach
- Event Sequences Involving Graphite Oxidation
- SST Conclusions

Potential Event Sequences Involving Graphite Oxidation – Addressing SRM 93-092

- SRM 93-092 – “The Commission believes that, for the MHTGR, the staff should also address the following type of event. The loss of primary coolant pressure boundary integrity whereby air ingress could occur (from the "chimney effect") resulting in a graphite fire and the subsequent loss of integrity of the fuel particle coatings.”

Previous NRC Staff Positions for MHTGR Graphite Oxidation Event Sequences

- From (1989) NUREG-1338, Appendix B – Summary of BNL Independent Analysis in Support of Safety Evaluation Report, Section 3, Evaluation of Large Air Ingress Scenarios (p7) – "For the graphite oxidation to proceed to the point that structural damage inside the core would become possible, an unlimited air supply would have to be available for many days."

- From (1995) NUREG-1338 (p 3-15) – "The staff concluded in draft NUREG-1338 that a graphite fire in the MHTGR core is a very low probability event. As stated in NUREG/CR-6218 on air ingression during severe accidents, without two breaches of the reactor vessel to create a chimney effect, it is not likely that significant amounts of air will enter into the core….<u>Therefore, graphite fires are not a licensability issue for the MHTGR.</u>"

NGNP Approach to Event Sequences Involving Graphite Oxidation

- Consistent with the findings of NUREG-1338 and the ACRS, it is expected that the frequency of the event type described in SRM 93-092 will fall so far below the LBE-spectrum of events (well below 5×10^{-7} per plant year) that the event would be considered incredible
- These expectations will be confirmed during the design process, once additional design detail is available
- Physically plausible bounding event sequences that maximize the potential for graphite oxidation will be considered in the bounding event sequence process as part of the NGNP licensing effort
- AGR Fuel Development and Qualification Program will obtain more data on air (and moisture) ingress effects

Next Generation Nuclear Plant

- Requested NRC Staff Position Regarding Siting Source Terms
- NGNP Siting Source Terms Approach
- Event Sequences Involving Graphite Oxidation
- SST Conclusions

Next Generation Nuclear Plant

SST Conclusions

- The NGNP SSTs approach is essentially the same as that proposed by DOE in the MHTGR PSID and accepted by the NRC staff in NUREG-1338
- The approach is consistent with discussions of containment function and mechanistic source terms in more recent NRC SECY documents and with approaches previously reviewed by the NRC staff for modular HTGRs
- Limiting LBEs will be evaluated to determine SSTs
- Physically plausible Bounding Event Sequences, including those involving graphite oxidation, will be considered to ensure that there are no cliff edge effects

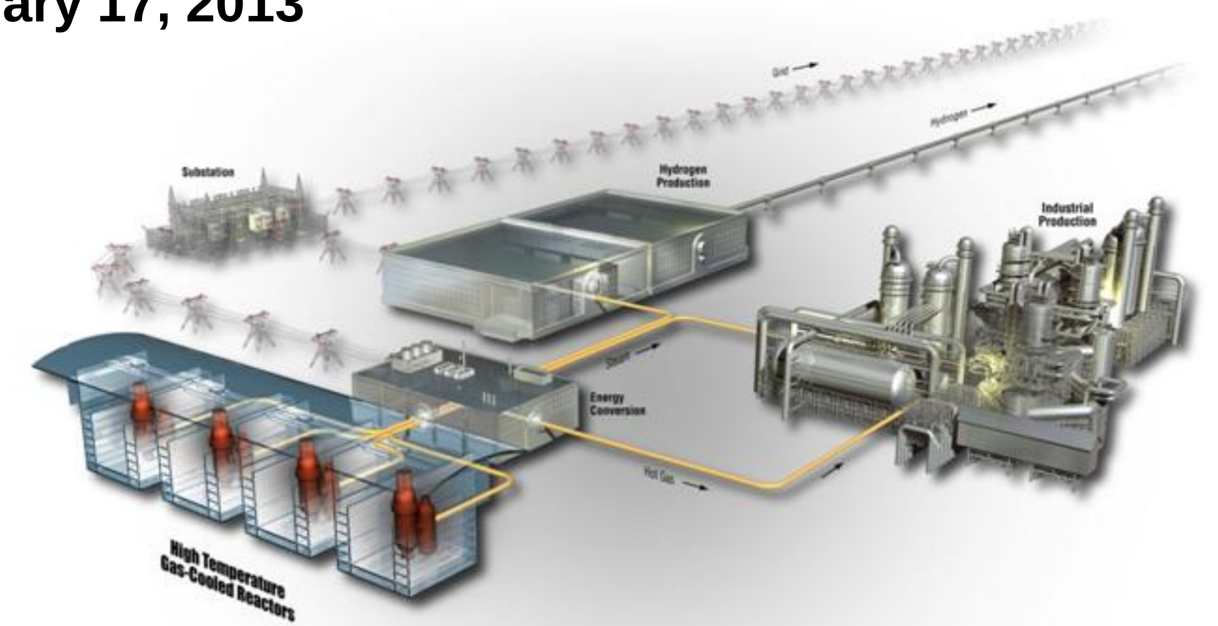
Requested NRC Staff Positions on Siting Source Terms – Recap

- Item 1.c. Establish a staff position to support a final determination regarding how LBEs will be considered for the purpose of plant siting and functional containment design decisions
  - In particular, we request that this staff position provide an adaptation of the guidance that has generally been applied to light water reactors (LWRs) for compliance with 10 CFR 100.21.

Fuel Qualification and Radionuclide Retention

ACRS Future Plant Designs Subcommittee Meeting

January 17, 2013



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Idaho National Laboratory

Meeting Agenda

- HTGR Safety Design Bases
- Licensing Basis Event (LBE) Selection Process
- Functional Containment Performance and Mechanistic Source Terms
- Siting Source Terms
- Fuel Qualification and Radionuclide Retention

Outline

- NGNP White Paper and Requested NRC Staff Positions
- Background
- Fuel Qualification Approach
  - Key Questions
- Fuel Qualification Program: Plans, Status and Key Results as they relate to Licensing
  - Fabrication
  - Fuel Irradiation
  - Fuel Post-Irradiation Examination
  - Fuel Safety Testing
- Summary and Path Forward

White Paper

Next Generation Nuclear Plant Fuel Qualification White Paper INL/EXT-10-18610

July 2010

ML 102040261


INL/EXT-10-18610
Revision 0

NGNP Fuel Qualification
White Paper

July 2010

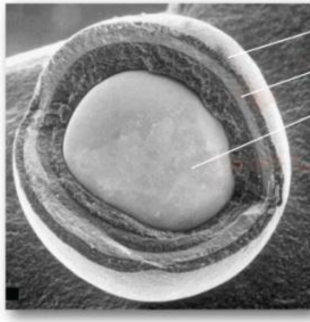
The INL is a U.S. Department of Energy National Laboratory operated by Battelle Energy Alliance.

Next Generation
Nuclear Plant

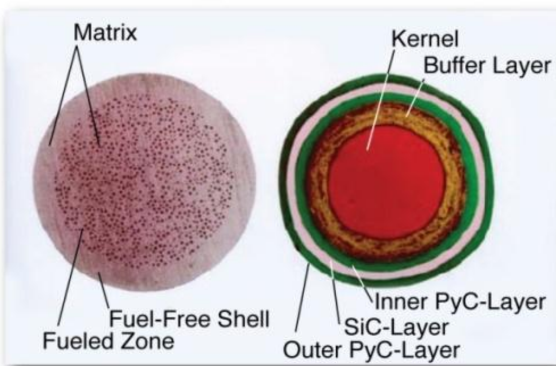
July 6, 2012 NGNP Letter to Staff: Requested NRC Staff Positions on Fuel Qualification

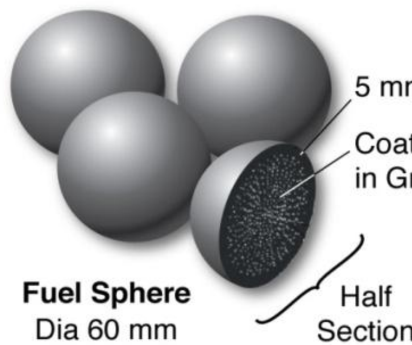
Item 1.a: Confirm plans being implemented by the Advanced Gas Reactor Fuel Development and Qualification Program are generally acceptable and provide reasonable assurance of the capability of coated particle fuel to retain fission products in a controlled and predictable manner. Identify any additional information or testing needed to provide adequate assurance of this capability, if required.

TRISO Fuel



TRISO-coated fuel particles (left) are formed into fuel compacts (center) and inserted into graphite fuel elements (right) for the prismatic reactor





TRISO-coated fuel particles are formed into fuel spheres for pebble bed reactor

08-GA50711-01

Introduction

- Fuel's ability to retain fission products is extremely important to the safety case and licensing approach for modular HTGRs. Key principles for this fuel:
  - High quality, low defect TRISO fuel can be fabricated and characterized in a repeatable and consistent manner
  - Fuel performance with very low in-service failures is achievable within anticipated modular HTGR fuel design envelope and can be calculated to the requisite level of accuracy
- UCO is the fuel form being qualified
  - UCO a mixture of UO_2, UC, and UC_2
  - Enables better fuel performance at higher burnup than UO_2 TRISO
  - UCO designed to provide excellent fuel performance at high burnup
    - Kernel migration suppressed (most important for prismatic designs because of larger thermal gradients)
    - Minimizes CO formation; internal gas pressure reduced
    - Fission products largely immobilized as oxides
    - Allows longer, more economical fuel cycle

Approach to NGNP Fuel Qualification

- Establishment of a fuel product specification (kernels, coatings, compacts)
- Implementation of a fuel fabrication process capable of meeting the specification
- Implementation of statistical quality control procedures to demonstrate that the specification has been met
- Irradiation of statistically sufficient quantities of fuel with monitoring of in-pile performance and post-irradiation examination to demonstrate that normal operational performance requirements are met
- Safety testing of statistically sufficient quantities of fuel to demonstrate that accident condition performance requirements are met
- Data from the program are used to develop/improve and qualify models to predict fuel performance and fission product transport in the reactor

Key Questions to be Addressed in Fuel Qualification

- What are the reactor designer's quality and performance requirements for fuel?
- Can the fabrication process meet those requirements?
- Will UCO TRISO fuel be able to meet the performance requirements under normal operating conditions?
- Will UCO TRISO fuel be able to meet the performance requirements under accident conditions?
- How well do representative models predict what is being observed?
- What else have we learned about fuel behavior and fission product transport?

Answers based on results to date provided in red font in presentation

Nominal Maximum Service Conditions (Based on Historical MHTGR Designs)

Parameter	Maximum Target Value
Peak Fuel temperature – normal operation, °C	1,400
Maximum time averaged fuel temperature (normal conditions), °C	1,250
Peak Fuel temperature (accident conditions), °C	1,600
Fuel burnup, % FIMA	18[a]
Fast fluence, 10^{25} n/m^2 (E > 0.18 MeV)	5
a. Estimated value for 15.5% enriched 425-μm reference fuel particle.	

Preliminary Fuel Quality and Performance Requirements (Based on Historical MHTGR Designs)

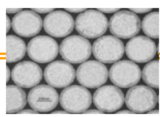
Parameter	NGNP – 750°C Core Outlet Temperature	
	Maximum Expected (Mean Value)	Design (95% confidence Value)
As-Manufactured Fuel Quality		
HM contamination	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Missing or defective buffer	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Missing or defective IPyC	$\leq 4.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
Defective SiC	$\leq 5.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
Missing or defective OPyC	0.01	0.02
In-Service Fuel Failure		
Normal operation	$\leq 5.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-4}$
Core heat-up accidents	$\leq 1.5 \times 10^{-4}$	$\leq 6.0 \times 10^{-4}$

Key for source term analysis

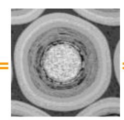
Scaling Up Kernel Production, Coating, Overcoating and Compacting Processes to Create a Pilot Line

Lab Scale

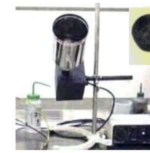
Sol-Gel Kernel Production



Lab Scale 2 inch CVD Coating (60 g charge)



Prepare Matrix



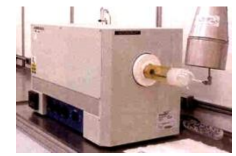
Overcoat and Dry

Sieve

Table

Riffle

Compact



Carbonize

Heat Treat

Engineering Scale

Kernel Forming and Drying

Industrial Scale 6 inch CVD Coating (2 kg charge)

Dry Mix and Jet Mill Matrix

Granurex Overcoat and Dry

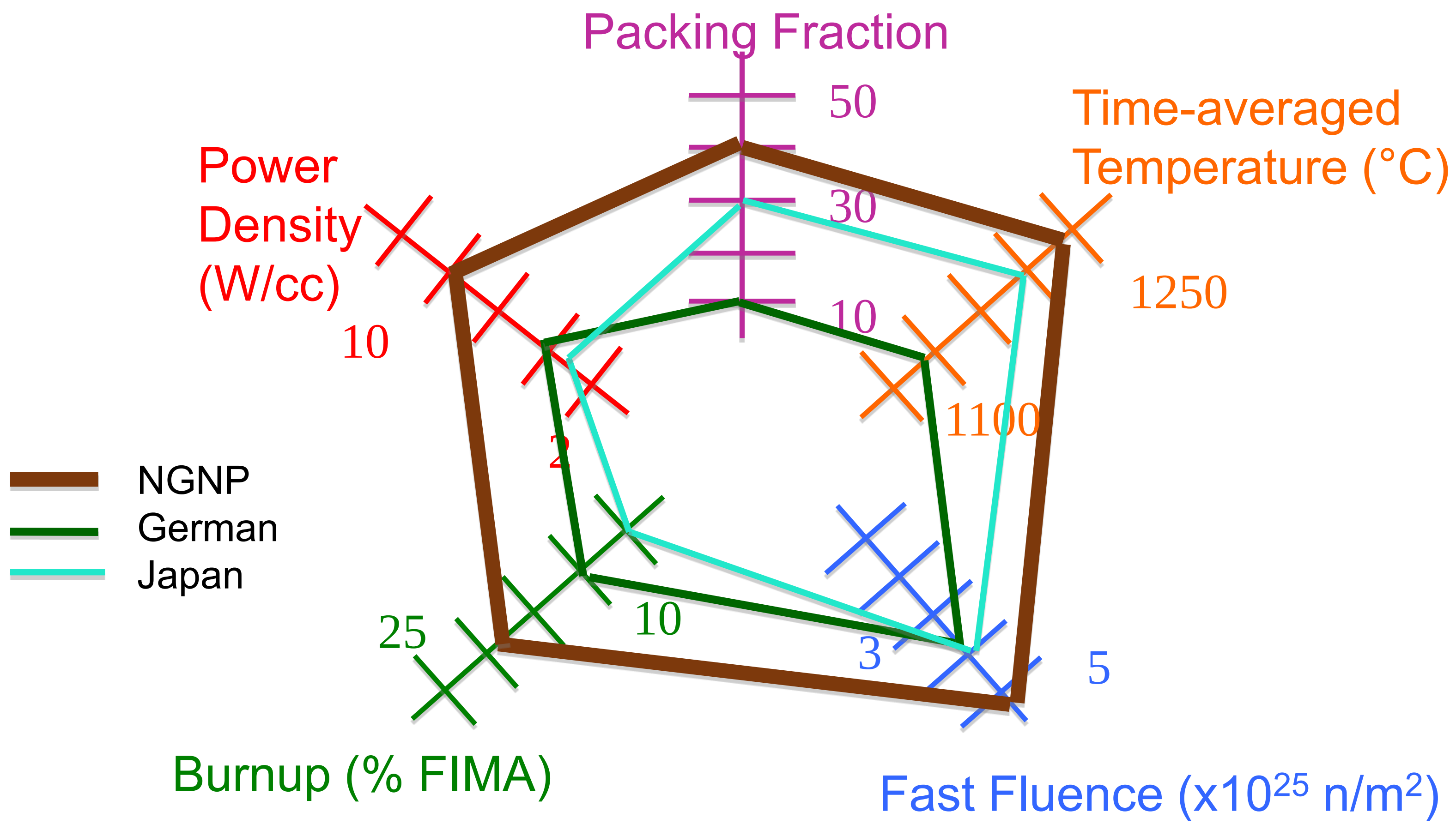
Hot Press Compact

Carbonize + Heat Treat in one Sequential Process

Fuel Fabrication Accomplishments

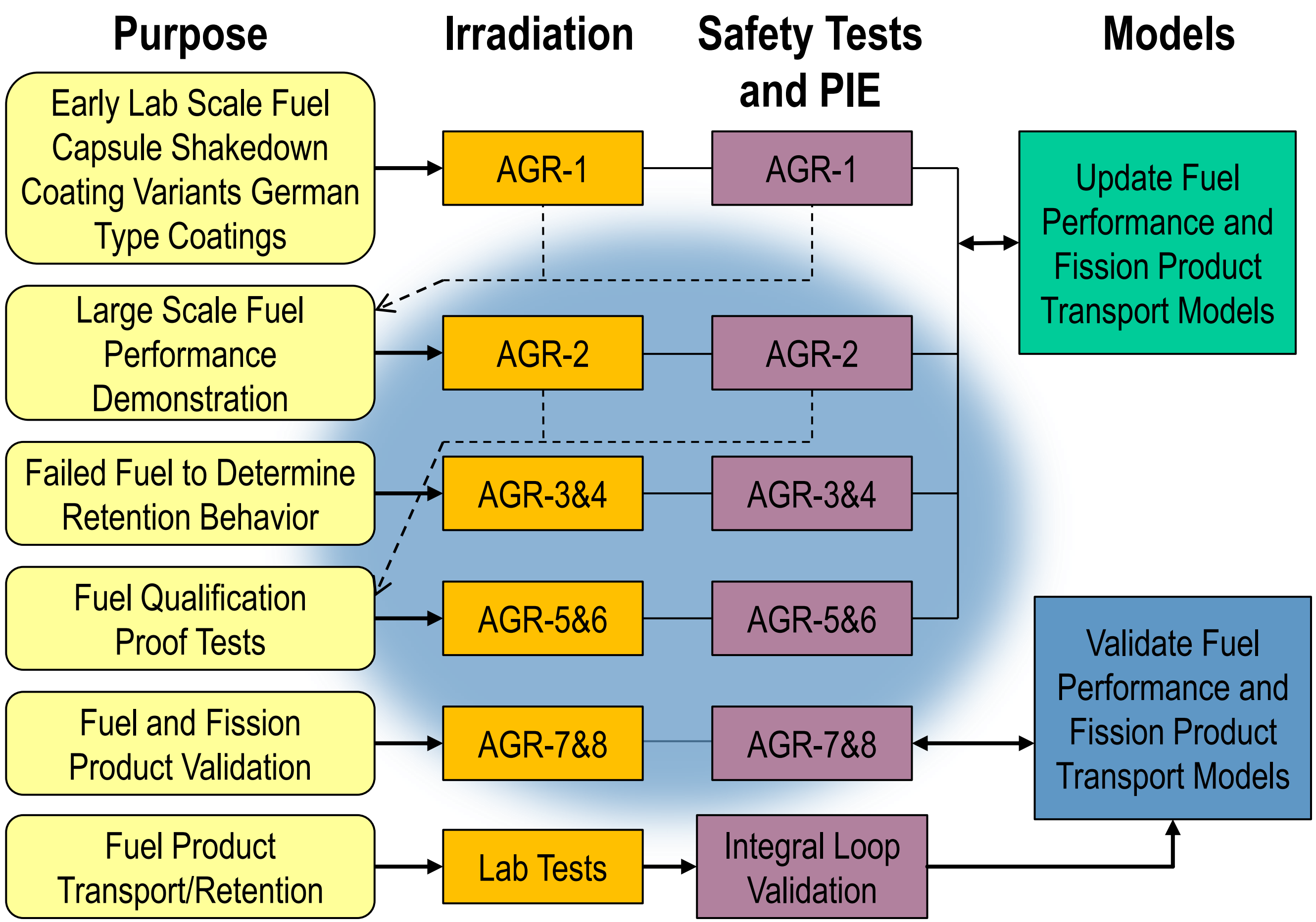
- Re-established capability to fabricate and characterize TRISO-coated particle fuel in the U.S. after a 10-15 year hiatus
- Developed a significantly improved understanding of how to fabricate high-performing TRISO fuel providing the technical basis for co-location of NGNP in industrial complexes
- Currently fabricating high-quality, low-defect TRISO-coated fuel particles in industry (B&W). Can meet physical specifications and are almost meeting all defect specifications at 95% confidence. With larger sample sizes and a mature process, should meet the defect specifications in production mode
- Vastly improved quality, reproducibility, process control, and characterization of TRISO fuel. Better control of the process, removal of high variability human interactions in the process, and better measurement technologies all contribute to better quality TRISO fuel
- Establishing a domestic vendor and associated fundamental understanding of key fuel fabrication parameters establishes credibility that the historical industrial experience from Germany in the 1980s is repeatable and has a sound technical basis
- All technologies needed to establish a pilot line are in industrial hands. Qualification fuel for AGR-5/6/7 will be produced in 2013

Performance Envelope for NGNP TRISO Fuel is more Aggressive than previous German and Japanese Fuel Qualification Efforts



Radar plot of five key parameters of fuel performance

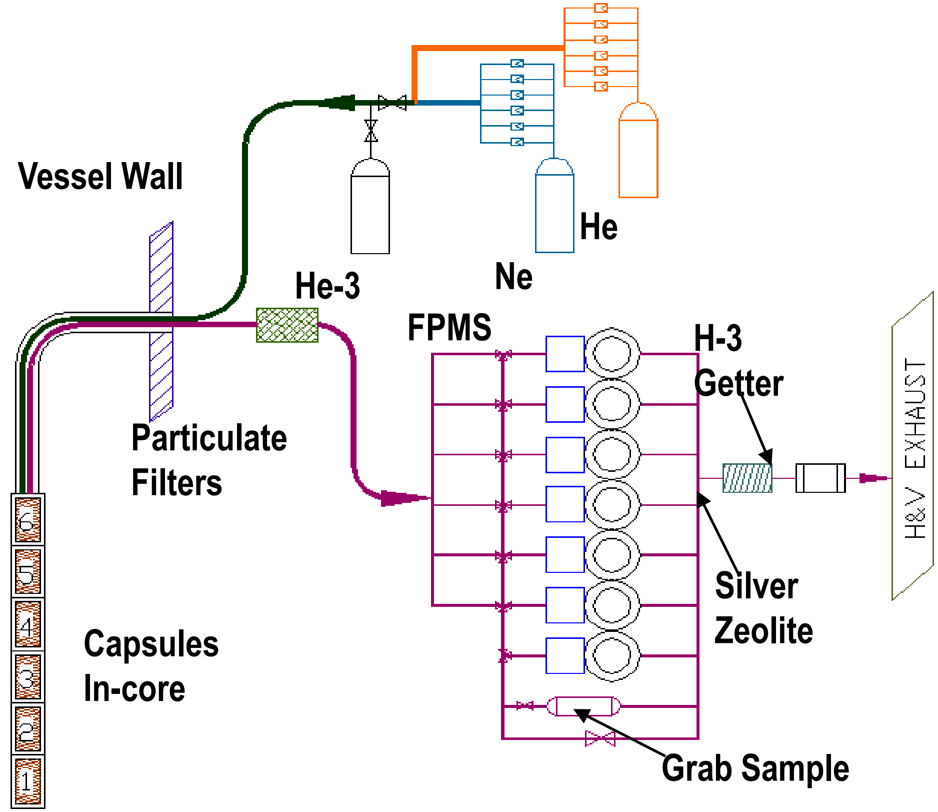
Overview of AGR Program Activities



Moisture and air ingress effects are part of AGR-5/6 PIE

NGNP Fuel Irradiation Capsule AGR-1 Demonstrated Outstanding Performance

- 350 um UCO TRISO; 19.7% enriched
- Goal burnup ~18–19% FIMA
- $\langle T \rangle_{max}$ <1250°C, $\langle T \rangle_{avg}$ ~1150°C
- Fast fluence <5×10^{25} n/m^2
- Irradiation began in December 2006 and completed November 2009
- *Peak burnup of 19% FIMA with no failures out of 300,000 particles*



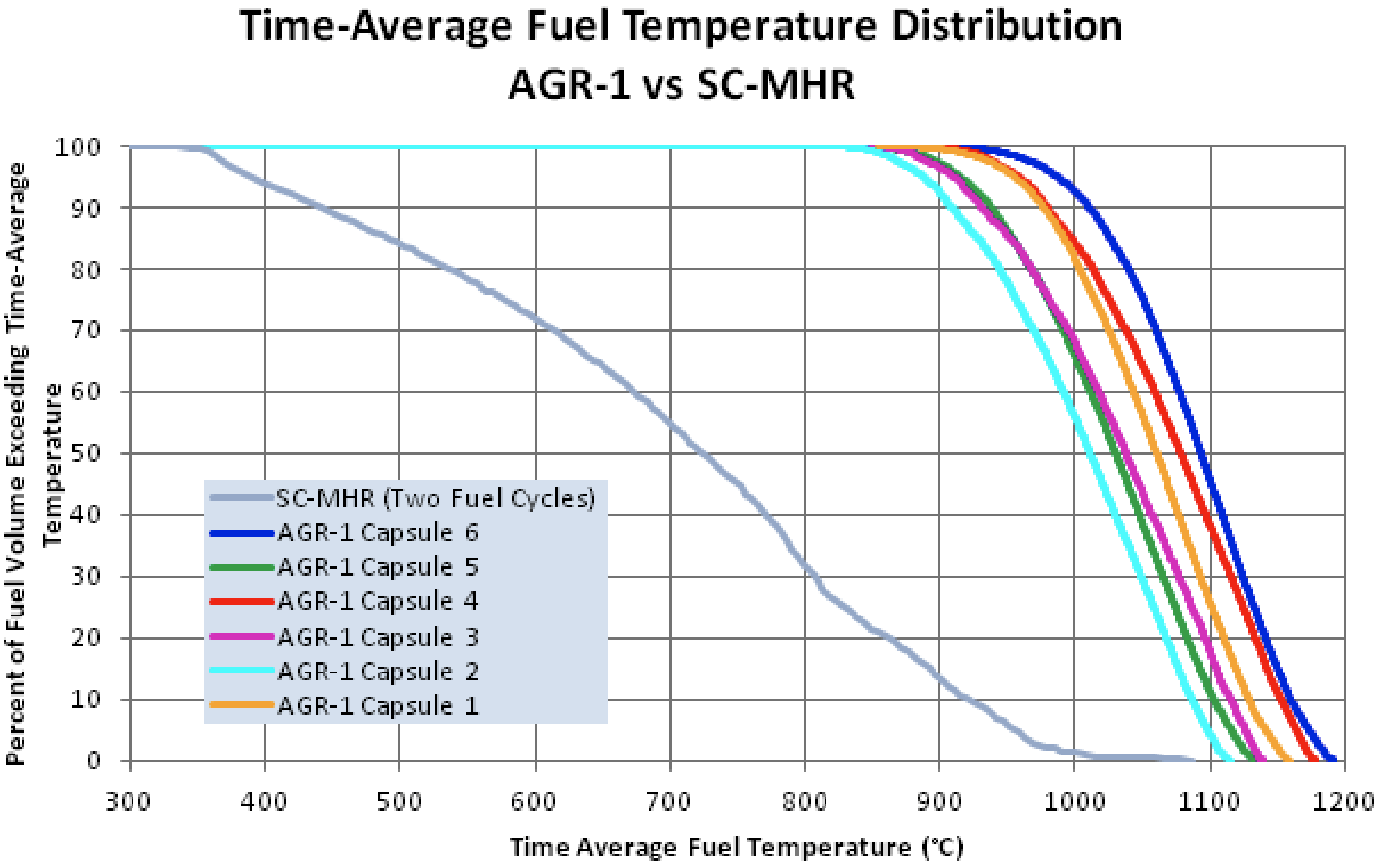
Individual capsule assembly with fuel compacts

Completed test train

Insertion into INL ATR

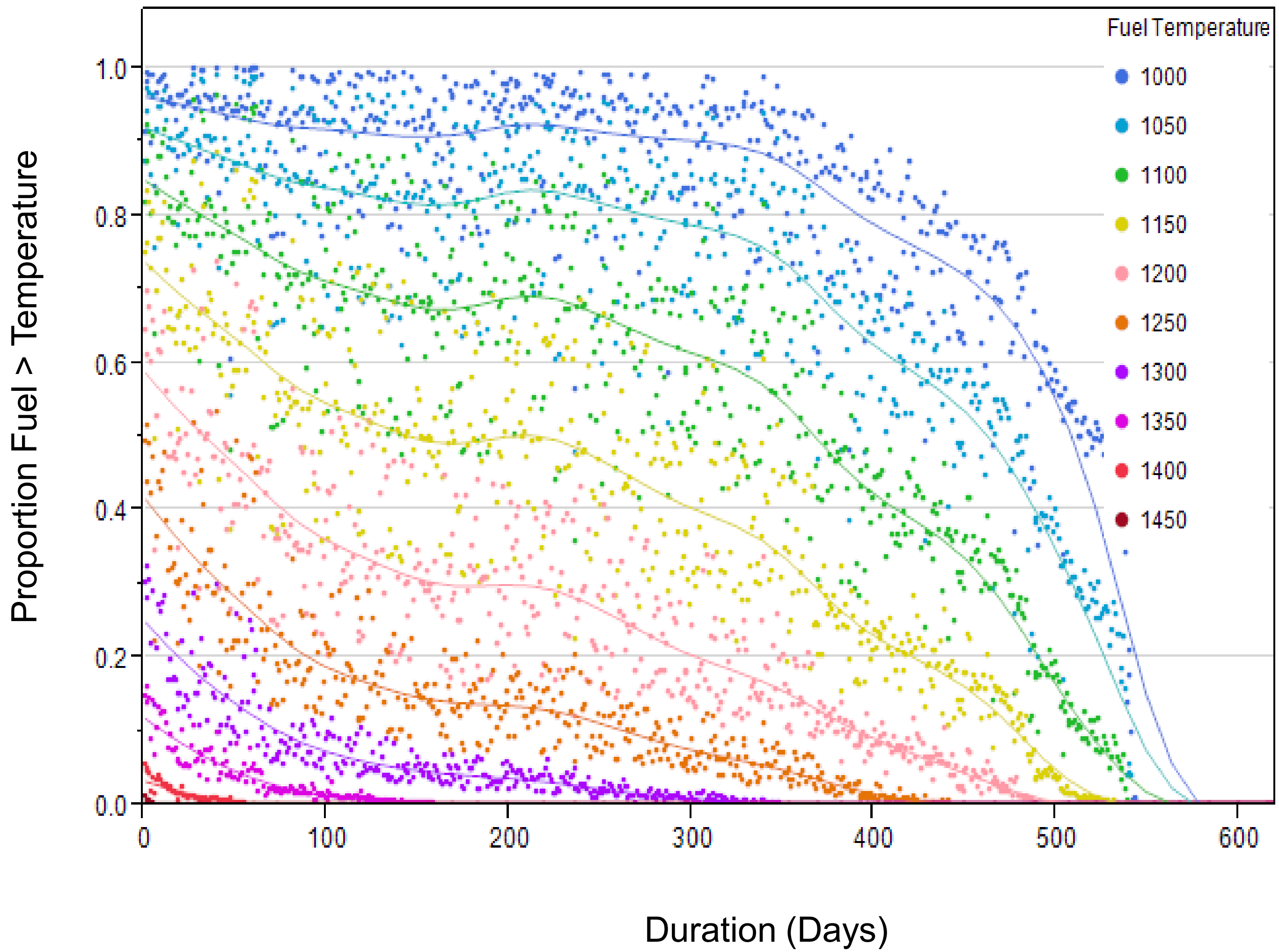
FPMS system

A wide range of temperatures, burnups and fluences were experienced by AGR-1 fuel compacts. Temperatures bound that expected in reactor.



SC-MHR is General Atomics conceptual design for NGNP

Large Quantities of TRISO Fuel Particles in the AGR-1 Irradiation Spent Significant Time at High Temperature



AGR-2 is Testing Vendor-produced UO_2 and UCO TRISO Fuel

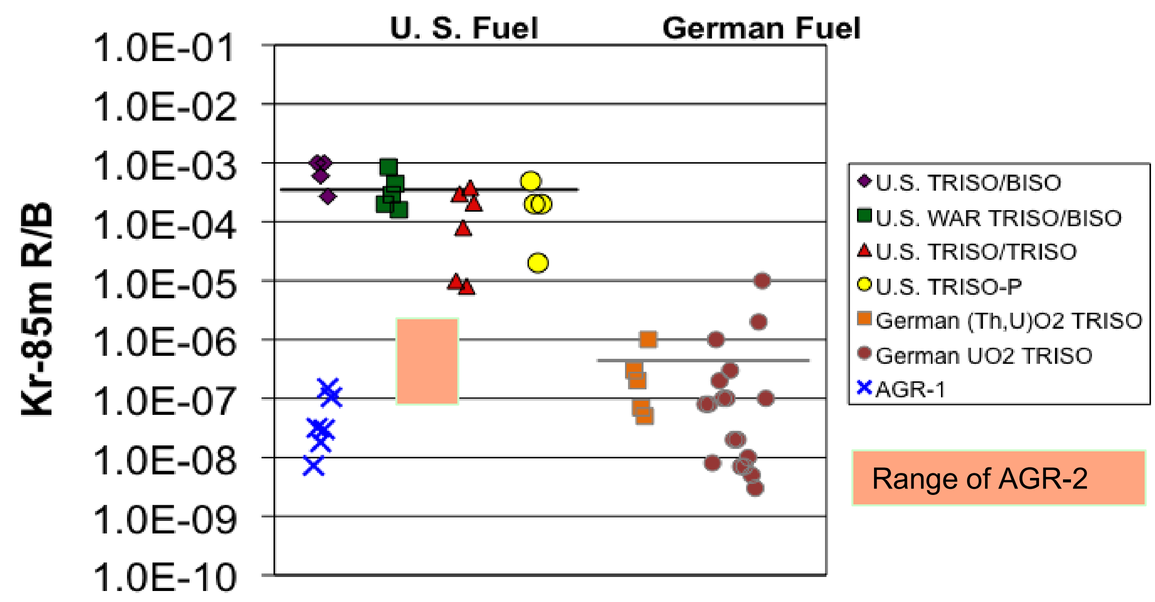
Capsule	Fuel Type	Vendor	Enrichment	Peak Burnup Goal	Time-Average Peak Temperature
6	425 μm UCO	B&W	14%	12% FIMA	<1250°C
5	425 μm UCO	B&W	14%	14% FIMA	<1250°C
4	500 μm UO_2	PBMR	9.6%	11% FIMA	<1150°C
3	500 μm UO_2	B&W	9.6%	11% FIMA	<1150°C
2	425 μm UCO	B&W	14%	14% FIMA	<1400°C
1	500 μm UO_2	CEA	19.6%	16% FIMA	<1150°C

Irradiation began in June 2010. Expected to complete in September 2013

Next Generation Nuclear Plant

AGR-1 and AGR-2 TRISO Fuel R/B Results Demonstrate Excellent Fuel Performance

Release-to-birth ratio (R/B) is measure of gas release from the fuel and a direct indicator of fuel performance



	U.S.	German
Irradiation temperature (C)	930 - 1350	800 - 1320
Burnup (%FIMA)	6.3 - 80	7.5 - 15.6
Fast fluence (10^{25} n/m^2)	2.0 - 10.2	0.1 - 8.5

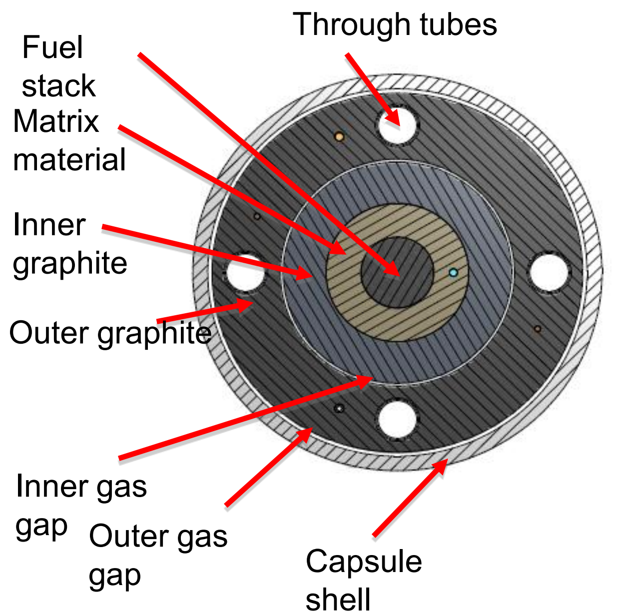
Fission Product Transport: Supporting a Mechanistic Source Term

- NGNP will use a mechanistic source term that takes into account the performance of all fission product release barriers (kernels, coatings, compact matrix, graphite, helium pressure boundary, reactor building) to meet radionuclide control requirements
- Goal is to provide technical basis for mechanistic source terms under normal and accident conditions to support reactor design and licensing
- Experimental data to be generated by irradiation experiments (AGR-3/4 and 8), PIE, safety testing, and loop testing
- Independent validation experiments are part of the plan

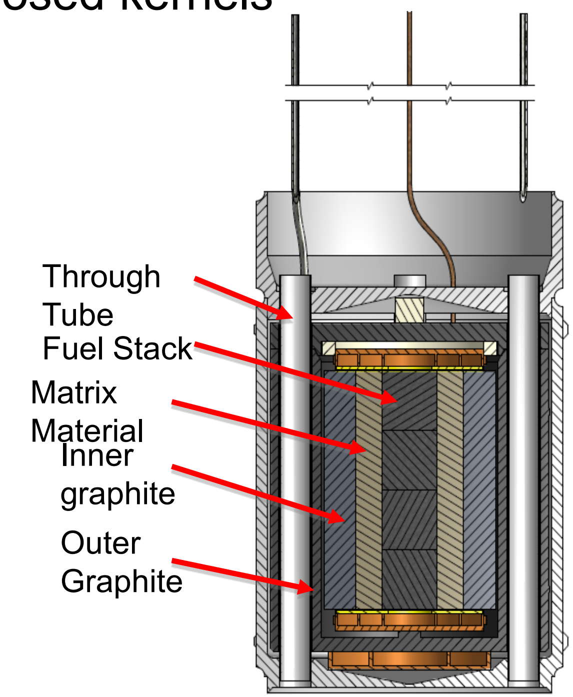
AGR-3/4 has Designed-to-fail Fuel and is Performing as Expected

- Need to understand the behavior of fission products released from the small fraction (~10^{-5}) of defective fuel and retained in graphitic components in the core
- Use “designed-to-fail” fuel that will provide a known source of fission product release
- Determine release rates of radionuclides from exposed kernels
- Establish metallic fission product transport and retention in fuel matrix and fuel element graphite

- Twelve separate capsules to span the temperature, burnup, and fluence envelope



AGR-3/4 Capsule Cross Section



Axial Cutaway of One of the 12 Capsules

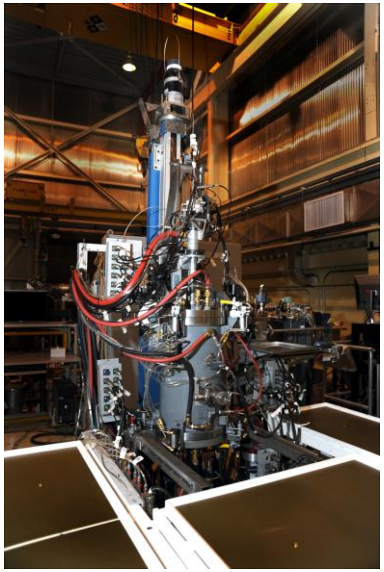
TRISO Fuel Irradiation Qualification Accomplishments

- Completed most successful U.S. irradiation of TRISO-coated particle fuel (AGR-1). 300,000 particles tested to peak burnup of 19.4% FIMA, a peak fast fuel of 4.5×10^{25} n/m^2 and peak time-average peak temperatures of 1250°C (peak MHTGR service conditions) with no failures
  - The expected superior irradiation performance of UCO at high burnup has been confirmed - no kernel migration, no evidence of CO attack of SiC, and no indication of SiC attack by lanthanides
- The AGR-1 95% confidence failure fraction is <1E-5, a factor of 20 better than the design in-service failure fraction of 2E-4. The more severe AGR-1 irradiation conditions compared to the vast majority of historic modular HTGR designs suggest substantial fuel performance margin
- Irradiation of AGR-2 is underway; no failures to date. Completion in September 2013
- Irradiation of AGR-3/4 is underway to study release/retention of fission products from failed TRISO fuel. Will complete in April 2014. This experiment will provide data needed for source term evaluations for UCO TRISO fuel, new matrix and graphite

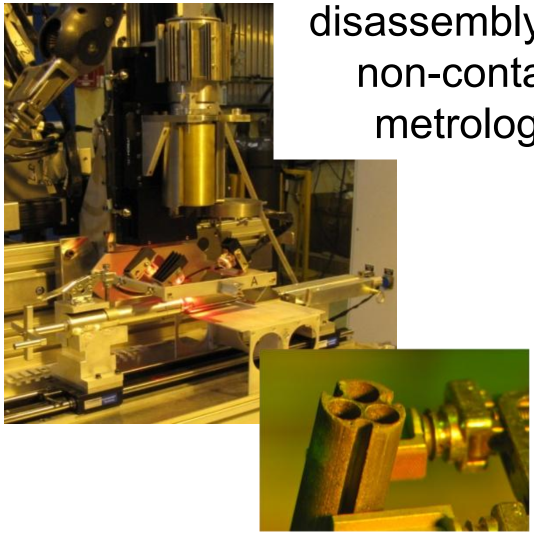
Post-Irradiation Examination Activities

- Infrastructure to meet fuel objectives is largely in place
- Objectives:
  - Detailed characterization of fuel behavior after irradiation in the reactor
  - Mass balance of fission products is critical for reactor source term
  - High temperature safety testing is required to establish fuel behavior under accident conditions

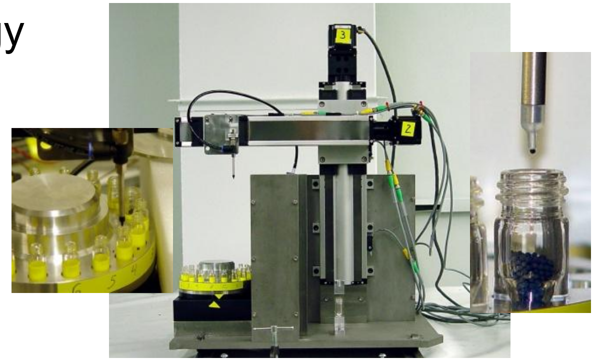
ORNL Furnace



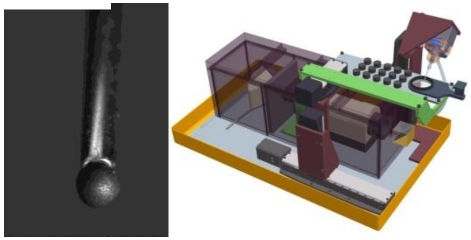
INL Furnace



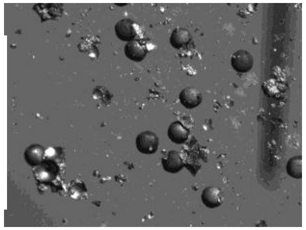
Capsule disassembly and non-contact metrology



Advanced-IMGA



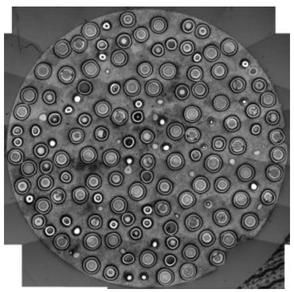
Particle handling and inspection



Deconsolidated AGR-1 particles and matrix

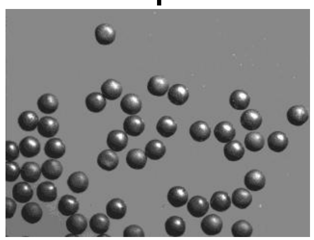
Using advanced characterization techniques to characterize fuel and fission product interactions from the millimeter to nanometer scale is improving our understanding of TRISO fuel behavior

10 mm



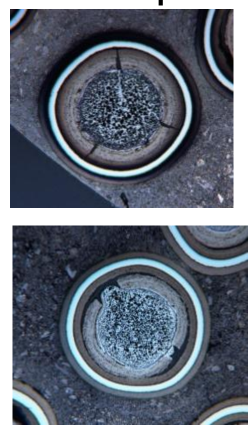
Cross-section of Compact

500 µm



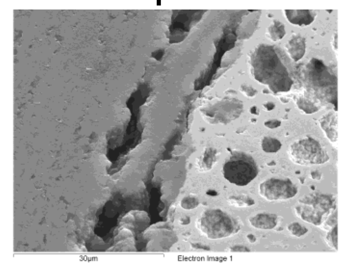
Individual Particles

100 µm



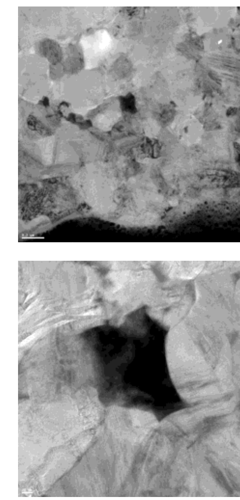
Micrograph of Particles

1 µm



SEM Image of Buffer Kernel Interface in Particles

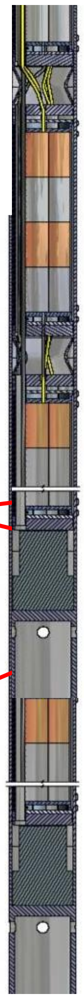
1 nm



High Resolution FIB/TEM Images of Precipitate near IPyC/SiC Interface

Methodology for isolating and identifying failed TRISO fuel particles greatly improves ability to characterize and understand fuel performance

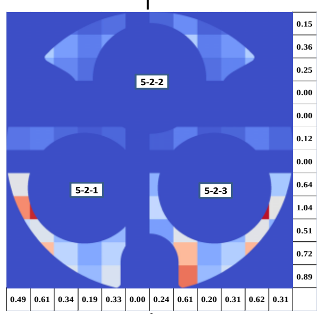
300,000 particles in irradiation



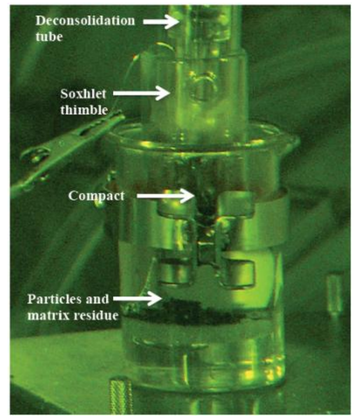
Fuel Compacts

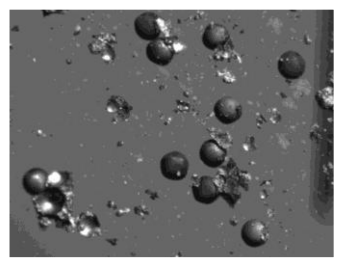
Plenum between Capsules

AGR-1 Test Train Vertical Section

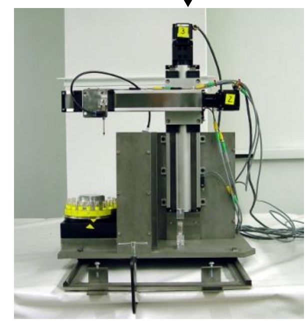


Gamma scan tomography to identify cesium hot spot and compact location

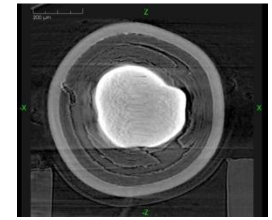


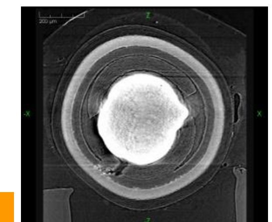


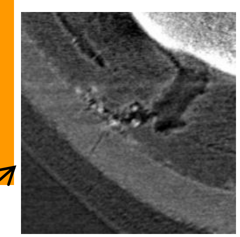
Deconsolidation to obtain particles from compact

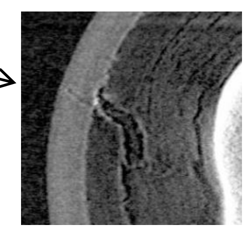


IMGA to find particles with low cesium retention









X-ray tomography and detailed micro-analytic techniques to study nature of defect

We know how many defective particles were in AGR-1, and have found them all and characterized them

NRC Follow-up Item: ATR irradiations do not produce enough Pu (and thus Pd) relative to that in the reactor and Pd corrosion of SiC at high burnup in TRISO fuel is an important degradation mechanism

- Much more Pd produced from Pu fissions than U fissions. Important in high burnup LEU fuel
- Concentration of Ag and Pd produced in AGR-1 during ATR irradiation is about 40 and 33% respectively below that expected in a prismatic HTGR at a peak burnup of ~20% FIMA
- FTE-13, a test of PuO_2 TRISO fuel (Peach Bottom), was irradiated to 70% FIMA and typical HTGR temperatures and levels of radiation damage. Some Pd interaction with SiC was observed, but no large-scale degradation of the SiC layer was observed.
- Volumetric Pd concentration in FTE-13 with PuO_2 kernels is 75× that of AGR-1. Areal concentration is 60×. Concerns raised by NRC about Pd attack are not expected to be significant under NGNP irradiation conditions
- The concentration differences in AGR irradiations and the HTGR are small compared to the level of Pd generated in FTE-13. AGR-1 PIE is providing new understanding of Pd interactions with SiC that suggests Pd is less of an issue than previously thought

NRC Follow-up item: Will high Ag or Pd release cause degradation of SiC and allow Cs release from the particles? Is there an enhancement to cesium diffusion under irradiation?

- No evidence of Ag and/or Pd affecting Cs release under normal or accident conditions in German database
- No evidence that Ag or Pd release affects Cs release from AGR-1
  - Large Ag release in AGR-1, but no cesium release
  - ~ 1% of Pd is outside the SiC in AGR-1, but SiC layer is retentive of Cs and no "attack" has been observed
  - Minimal release of Cs to the matrix implies no substantial Pd degradation of SiC layer
  - No release of Cs in AGR heating tests (at 1600 and 1700°C) to date unless compact had an SiC defect
- Minimal release of Cs to the matrix implies no enhanced diffusion under irradiation. Low releases to the matrix suggest IAEA TECDOC diffusion coefficient at normal operating temperatures is conservative

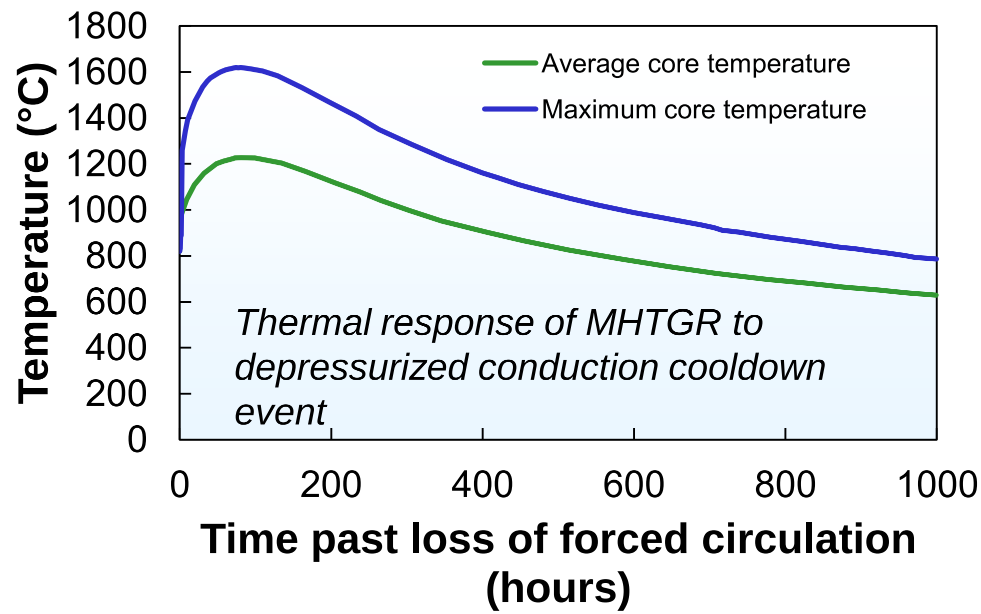
- AGR-1 data show no evidence of degradation or enhancements effects influencing cesium release/diffusion

TRISO Fuel Post-Irradiation Examination (PIE) Accomplishments

- Post-Irradiation examination is revealing new understanding of fuel performance and fission product transport
  - Characterization of kernel and coating behaviors to better understand performance and potential failure modes
  - More complete mass balance of key fission products (Ag, Cs, Sr, Eu, Ce, Pd)
    - IMGA to examine particle to particle variability and to identify defective particles that release fission products
    - Gamma scanning of test train components and deconsolidation of fuel compacts to evaluate retentiveness of SiC layer
  - Fission product/SiC interactions
    - Characterizing fuel and coating layer microstructures at micro and nano-scale
    - No Pd corrosion or attack of SiC has been observed! Models overpredict SiC corrosion by Pd
    - Models conservatively overpredict release of Cs from fuel under normal operation

Accident Safety Testing of TRISO Fuel

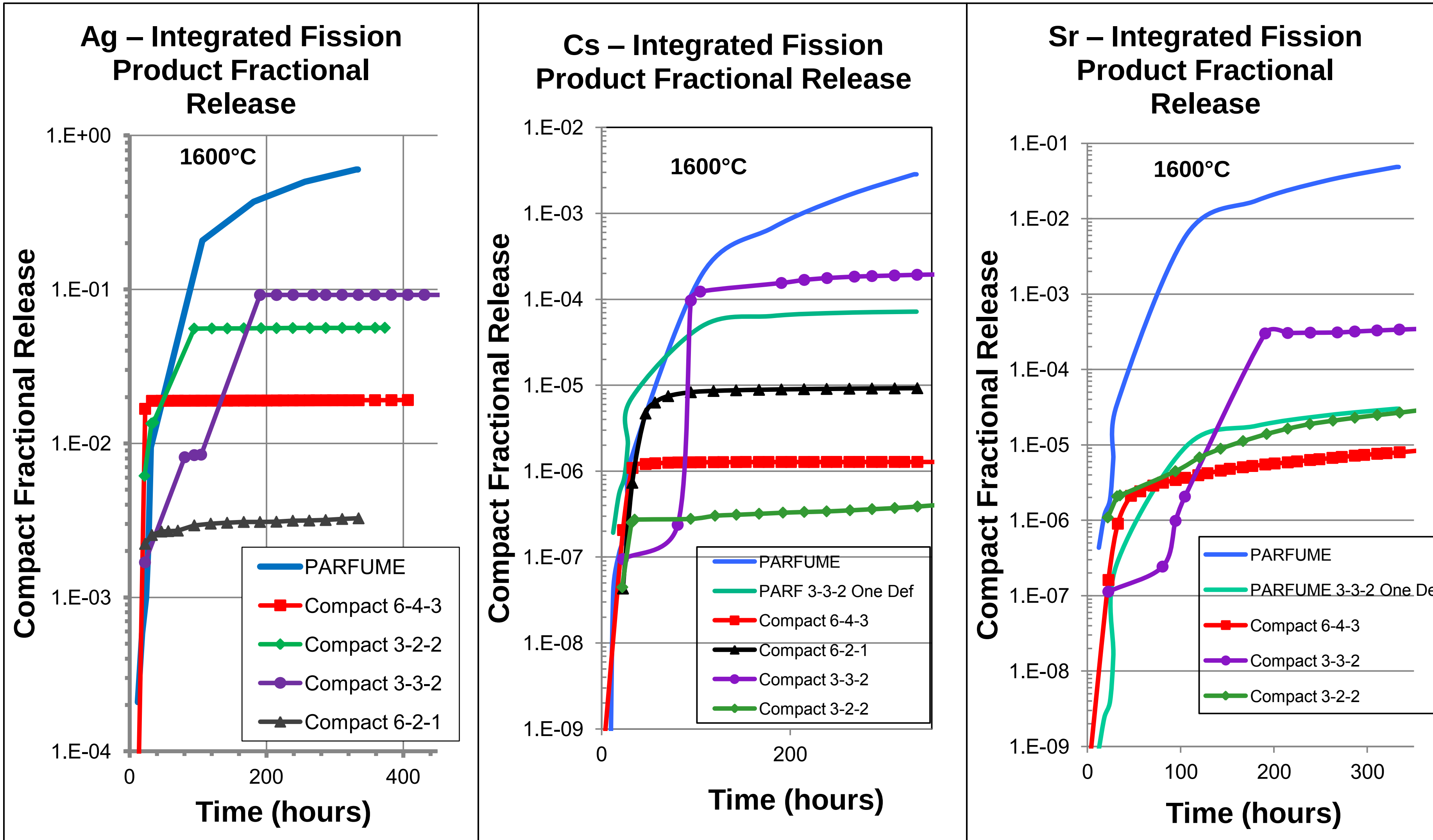
- Simulate heatup of fuel compacts following depressurized conduction cooldown event
- Isothermal testing for hundreds of hours at 1600, 1700, and 1800°C
- Five isothermal 1600 and 1700°C tests have been completed
- An 1800°C isothermal test will be performed this year
- Actual time-temperature test to be performed this year
- Testing of deconsolidated particles will occur in late 2013 or early 2014



Key Results

- Releases are very low unless a defective particle is present
- Releases are from fission products that diffused into the matrix during the irradiation and not from the intact TRISO particles during the high temperature heating

Historical IAEA TECDOC data used in models significantly overpredicts release under AGR isothermal heating tests



TRISO Fuel Safety Testing Accomplishments

- Accident safety testing of UCO TRISO from AGR-1 is nearing completion and demonstrating robustness of fuel
  - Very low releases after hundreds of hours at 1600 and 1700°C. No particle failures (no noble gas release measured)
  - Releases are associated with fission products that diffused into the matrix during the irradiation. No diffusive release from intact TRISO particles during the high temperature heating
  - UCO TRISO fuel should be able meet in-service failure fraction under off-normal conditions, but more data are needed to demonstrate statistical significance. Still need data on performance under water and air ingress conditions
  - Historical database of diffusion coefficients significantly overpredicts measured releases

Path Forward

- Complete safety testing and PIE on AGR-1 fuel in 2013 (including safety testing at 1800°C)
- Complete AGR-2 performance demonstration irradiation in 2013
- Complete AGR-3/4 source term irradiation in 2014
- Perform PIE and safety testing of AGR-2 and AGR-3/4 in 2014-2015
- Fuel qualification and margin irradiation (AGR-5/6/7) is scheduled for 2016
- Moisture and air effects on fuel are scheduled as part of PIE campaign for AGR-5/6/7 in 2018-2020
- AGR-8 will follow AGR-5/6/7

Key Results of On-going Research

Fabrication

- Improved understanding of TRISO fuel fabrication process
- Improved fabrication and characterization of TRISO fuel produced by fuel vendor

Irradiation Performance

- Outstanding irradiation performance of a large statistically significant population of TRISO fuel particles under high burnup, high temperature HTGR conditions
- Expected superior irradiation performance of UCO at high burnup has been confirmed

Post-Irradiation Examination and Safety Testing

- Post-Irradiation examination of AGR-1 indicates:
  - Ag release consistent with model predictions
  - No Cs release from intact particles under irradiation
  - No Pd attack or corrosion of SiC despite large amounts of Pd outside SiC
- Initial safety testing at 1600 and 1700°C demonstrating robustness of UCO TRISO under depressurized conduction cooldown conditions
  - Low releases from intact particles. Releases attributed to defective particles and transport of fission products released during irradiation. No particle failures observed to date

Summary

- The AGR Fuel Development and Qualification Program will provide data necessary to better understand fuel performance and fission product behavior for modular HTGRs
- The AGR Fuel Program is laying the technical foundation needed to qualify UCO TRISO fuel made to fuel process and product specifications within an envelope of operating and accident conditions that are expected to be bounding for modular HTGRs
- AGR results to date are consistent with current design assumptions about fuel performance and radionuclide retention. Program is obtaining additional data to support model development and validation
- AGR results to date are consistent with the safety design basis, including the functional containment and mechanistic source term approaches presented today

Requested NRC Staff Positions on Fuel Qualification – Recap

Item 1.a: Confirm plans being implemented by the Advanced Gas Reactor Fuel Development and Qualification Program are generally acceptable and provide reasonable assurance of the capability of coated particle fuel to retain fission products in a controlled and predictable manner. Identify any additional information or testing needed to provide adequate assurance of this capability, if required.

Significance of Graphite Oxidation to Public Safety

- Graphite will chemically react (oxidize) with oxygen in air or in a helium-air gas mixture
- Nuclear grade graphite is much less reactive than other types of graphite due to its graphitized structure and high purity
- Oxidation of graphite is limited by
  - the amount of air in the helium gas mixture from the reactor building
  - the high flow resistance of the coolant channels to the core height (L/D > 700)
- Fuel particles are embedded in the graphite matrix within the fuel element
- Loss of all forced cooling and depressurization of HPB required for air to ingress
- MHTGR analyses for an assumed large HPB failure of 22 sq ft showed only 1% of core graphite oxidized after 30 days with 8 RB volumes of 100% air ingressed
  - Oxidation resulted in small contribution to heat generation compared to decay heat
  - Oxidation did not lead to loss of core geometry
  - No appreciable incremental radionuclide release due to oxidation
- NGNP analyses have shown that a break in the HPB leads to a small percentage of air in the gas mixture after the helium blowdown