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

May 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, APRIL 9, 2013, ROCKVILLE, MARYLAND

The minutes for the subject meeting were certified on May 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, APRIL 9, 2013, ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on April 9, 2013, are an accurate record of the proceedings for that meeting.

__________***/RA/***__________________

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

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING MINUTES
APRIL 9, 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 April 9, 2013. The Subcommittee was briefed by representatives of the U.S. Department of Energy (DOE), Idaho National Laboratory (INL), and the NRC staff regarding NGNP research and key licensing issues pertaining to DOE's Next Generation Nuclear Plant (NGNP) Project. The INL presentations also included information regarding DOE/INL's defense-in-depth approach and reactor building design alternatives.

The meeting convened at 10:00 AM and adjourned at 5:16 PM. The meeting was open to the public. No written comments were received from members of the public related to this meeting. No oral comments were received from members of the public during this meeting.

ATTENDEES

ACRS Members	Jonathon DeGange, NRC/NRO	Sardar Ahmed, NRO
Dennis Bley (Chairman)	Thomas Boyle, NRC/NRO	John McKirgan, NRO
Gordon Skillman	Jim Shea, NRC/NRO	Varoujan Kalikian, NRR
John Stetkar	Arlon Costa, NRC/NRO	Don Brittner, NRR
Michael Corradini	Michelle Hart, NRC/NRO	**Other Attendees**
Harold Ray	Michael Mayfield, NRC/NRO	David Hanson, INL
Joy Rempe	Mike Kania, NRC Consultant*	James Kinsey, INL
Thomas Kress (Consultant)	**NRC Staff**	Farshid Shahrokhi, AVEVA
ACRS Staff	Neil Ray, NRO	Tom O'Connor, DOE
Mark Banks (DFO)	Kimberly Gambone, NSIR	Janelle Zamore, DOE
Presenters	Cameron Goodwin, NRO	Thomas Hicks, INL
Carl Sink, DOE	George Thomas, NRO	John Kelly, DOE
Fred Silady, INL	Vanice Perin, NRO	Madeline Feltus, DOE
David Petti, INL	Russell Chazell, NRO	David Hanson, INL
David Alberstein, INL	Stu Magruder, NRO	Stuart Rubin, NUMARC
Mark Holbrook, INL	Sud Basu, RES	Patrick Troy, Lockheed-Martin
Anna Bradford, NRC/NRO	Matthew Humberstone, NRO	
Don Carlson, NRC/NRO	Courtney St. Peters, NRO	

* Participating by telephone

SUMMARY

The purpose of this meeting was for the ACRS Future Plant Designs Subcommittee to receive an information briefing from the NRC staff regarding its assessment of DOE/INL's positions related to NGNP key licensing issues:

- Licensing basis event selection
- Source terms
- Functional containment performance
- Emergency preparedness.

The Subcommittee also received a briefing from DOE and its lead laboratory, INL, on the NGNP Project. The DOE/INL briefing reviewed key messages from the DOE/INL January 17, 2013 Subcommittee briefing, discussed the NGNP defense-in-depth approach, and described reactor building design alternatives.

The staff viewed the DOE/INL's proposed approaches to the above NGNP key licensing issues as being generally reasonable. In regards to areas of staff concern, the staff believed that deterministic elements of licensing basis event selection should be strengthened and technical issues should be resolved through prototype testing in accordance with 10 CFR 50.43(e)(2). The staff also discussed potential policy issues which may require future Commission direction.

At the conclusion of the meeting, the Subcommittee members and their consultant commented on various aspects of the information presented by DOE/INL and the NRC staff. Several members expressed concern regarding the staff's de-emphasis of using probabilistic risk insights when determining design basis accidents. A member pointed out that there seemed to be a need of clarity between DOE/INL and the staff regarding the staff's intention to use the prototype regulation (10 CFR 50.34(e)(2)) to resolve outstanding design issues. A member mentioned the potential complexities related to operating multiple reactors from a common control room at a single site, as well as emergency planning challenges. A member expressed concern regarding the different treatments of uncertainty for anticipated events and beyond design basis events (mean), and design basis events (95%). In addition, there needed to be clarity on the definition of an event sequence and how event sequences will be used to identify and categorize licensing basis events. A member mentioned that the DOE/INL description of defense-in-depth appeared to be thorough. Also, the idea that changing the order of items in event trees could change the categorization needed additional consideration. The ACRS consultant agreed with many of the above observations and expressed concern regarding the lack of use of societal risk in the selection of accidents, the unresolved issue of air ingress, and issues with the top-level regulatory criteria (F-C curve): stair step versus vertical line and lack of how close is too close to a limit.

Finally, a table of significant issues discussed during the meeting is provided below, as a guide to the transcript.

DOCUMENTS PROVIDED TO THE COMMITTEE

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. Nuclear Regulatory Commission, SRM-SECY-08-0019, "Licensing and Regulatory Research Related to Advanced Nuclear Reactors," June 11, 2008 (ML081630507)
2. U.S. Nuclear Regulatory Commission, COMSECY-08-0018, "Report to Congress on Next Generation Nuclear Plant (NGNP) Licensing Strategy," May 12, 2008 (ML081330510)
3. U.S. Nuclear Regulatory Commission, SECY-11-052, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," October 28, 2011 (ML112570439)
4. Gibbs, G. A, Idaho National Laboratory, INL/EXT-11-22708, "Modular HTGR Safety Basis and Approach," August 2011 (ML11251A169)

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5. Idaho National Laboratory Letter, “Next Generation Nuclear Plant Submittal – Confirmation of Requested NRC Staff Positions,” July 6, 2012 (ML121910310)
6. Gibbs, G. A, Idaho National Laboratory, INL/EXT-10-17686, “NGNP Fuel Qualification White Paper,” July 2010 (ML102040261)
7. Gibbs, G. A, Idaho National Laboratory, INL/EXT-10-17997, “Mechanistic Source Terms White Paper,” July 2010 (ML102040260)
8. Gibbs, G. A, Idaho National Laboratory, INL/EXT-09-17139, “Next Generation Nuclear Plant Defense-in-Depth Approach,” December 2009 (ML093490191)
9. Gibbs, G. A, Idaho National Laboratory, INL/EXT-10-19521, “Next Generation Nuclear Plant Licensing Basis Event Selection White Paper,” September 2010 (ML102630246)
10. Gibbs, G. A, Idaho National Laboratory, INL/EXT-10-19509, “Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification White Paper,” September 2010 (ML102660144)
11. Gibbs, G. A, Idaho National Laboratory, INL/EXT-11-21270, “Next Generation Nuclear Plant Probabilistic Risk Assessment White Paper,” September 2011 (ML11265A082)
12. Gibbs, G. A, Idaho National Laboratory, INL/EXT-09-17187, “NGNP High Temperature Materials White Paper,” June 2010 (ML101800221)
13. Gibbs, G. A, 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. Nuclear Regulatory Commission, “Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms,” February 12, 2012 (ML120240669)
15. U.S. Nuclear Regulatory Commission, “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)
16. U.S. Nuclear Regulatory Commission, “Summary Feedback on Four Key Licensing Issues,” (draft), March 8, 2013 (ML13002A157)
17. U.S. Nuclear Regulatory Commission, “Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms,” Revision 1 (draft), March 11, 2013 (ML13002A168)
18. U.S. Nuclear Regulatory Commission, “Assessment of White Paper Submittals on Defense-in-Depth, Licensing Basis Event Selection, and Safety Classification of Structures, Systems, and Components,” Revision 1 (draft) March 8, 2013 (ML13002A162)
19. Gibbs, G. A, Idaho National Laboratory letter to U.S. Nuclear Regulatory Commission, “Contract No. DE-AC07-05ID14517 – Next Generation Nuclear Plant Project Submittal – Response to Nuclear Regulatory Commission Request for Additional Information Letter No. 002 Regarding Next Generation Nuclear Plant Project Fuel Qualification and Mechanistic Source Terms – NRC Project # 0748”, August 10, 2011 (ML11224A060)
20. Gibbs, G. A, Idaho National Laboratory letter to U.S. Nuclear Regulatory Commission, “Contract No. DE-AC07-05ID14517 – Next Generation Nuclear Plant Project Submittal – Response to Nuclear Regulatory Commission Request for Additional Information Letter No. 005 Regarding the Risk-Informed, Performance-Based Licensing Approach – NRC Project # 0748”, October 14, 2011 (ML11290A188)
21. Gibbs, G. A, Idaho National Laboratory letter to U.S. Nuclear Regulatory Commission, “Contract No. DE-AC07-05ID14517 – Next Generation Nuclear Plant Project Licensing White Paper – Next Generation Nuclear Plant Defense-in-Depth Approach – Response to

Nuclear Regulatory Commission Request for Additional Information Letter No. 001 – NRC Project # 0748”, September 15, 2010 (ML102590481)

22. Gibbs, G. A, Idaho National Laboratory letter to U.S. Nuclear Regulatory Commission, “Contract No. DE-AC07-05ID14517 – Next Generation Nuclear Plant Project Submittal – Response to Nuclear Regulatory Commission Request for Additional Information Letter No. 004 Regarding Next Generation Nuclear Plant Project High Temperature Materials White Paper – NRC Project # 0748”, September 27, 2011 (ML11272A067)
23. Gibbs, G. A, Idaho National Laboratory letter to U.S. Nuclear Regulatory Commission, “Contract No. DE-AC07-05ID14517 – Next Generation Nuclear Plant Project Submittal – Response to Nuclear Regulatory Commission Request for Additional Information Letter No. 003 Regarding Next Generation Nuclear Plant Project Fuel Qualification and Mechanistic Source Terms – NRC Project # 0748”, September 21, 2011 (ML11266A133)

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Significant issues from April 9, 2013 FPD SC meeting on NGNP

Issue descriptions below linked to location in attached transcript

SIGNIFICANT ISSUES	
Issue	Reference Pages in Transcript
DOE/INL	
Next NGNP Project direction according to DOE (Corradini)	8-11
Status of DOE/INL White Papers and Staff Assessments (Bley, Ray)	11-13
DOE/INL's meaning of licensing framework	13
Need for DOE/INL & NRC work to be clearly documented (Corradini)	13-14
Technology neutral framework (Corradini)	14-15
Discussion regarding relationship of DBEs and DBAs (Stetkar, Kress)	19-26
Event Sequence frequency and # of modules (Kress)	27-35
F-C curve x-axis uncertainty (Stetkar) – also addressed later	35-38
F-C curve stair step versus horizontal line cutoff (Kress)	38-40
Impact on analysis by differing # of units at a site (Rempe)	41-42
Lack of use of societal risk (Kress)	44-45
Uncertainty discussion (Stetkar)	46-49
Fuels discussion – fuel matrix versus particles releasing (Corradini)	57-58
DOE/INL plans to heat fuel particles in furnace	58
Thermocouple issues during fuel testing (Bley)	59-64
Discussion on no need for reactor building (DID) (Corradini)	67-68
Design to EPA PAGs versus 10 CFR 50.34 (DID) (Corradini)	68-70
Discussion on release related to dust and plate out (Corradini)	73-79
Discussion of releases from alternative considered reactor buildings (Bley)	81-94
Treatment of uncertainty inconsistency – mean vs. 95% (Stetkar)	106-109
Discussion on integrated defense-in-depth (Bley)	109-111
DOE/INL summary	112-114
NRC Staff	
Discussion on need for Commission policy decisions (Corradini)	129-131
Discussion on 10 CFR 50.43(e)(2) – prototype regulation	132-135
Discussion on what is a the definition of event sequence (Stetkar)	143-148
Staff Issue: DOE/INL event categorization (also discussion on probabilistic versus deterministic – Stetkar)	154-167
Staff Issue: specified acceptable fuel design limits (SAFDLs)	167
Staff Issue: DOE/INL approach regarding SSC classification	167-168

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards Future Plant Designs Subcommittee
Docket Number:	(n/a)
Location:	Rockville, Maryland
Date:	Tuesday, April 9, 2013

Work Order No.: NRC-4111 Pages 1-291

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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FUTURE PLANT DESIGNS SUBCOMMITTEE

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TUESDAY

APRIL 9, 2013

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

DENNIS C. BLEY, Chairman

MICHAEL L. CORRADINI, Member

HAROLD B. RAY, Member

JOY REMPE, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

ACRS CONSULTANT:

THOMAS S. KRESS

NRC STAFF PRESENT:

MARK BANKS, Designated Federal Official

ALSO PRESENT:

CARL SINK, DOE

FRED SILADY, INL

DAVID PETTI, INL

DAVID ALBERSTEIN, INL

MARK HOLBROOK, INL

JIM KINSEY, INL

ANNA BRADFORD, NRO

DON CARLSON, NRO

TOM BOYLE, NRO

JONATHAN DEGANGE, NRO

JIM SHEA, NRO

ARLON COSTA, NRO

MIKE MAYFIELD, NRO

MICHELLE HART, NRO

MIKE KANIA*

*Present via telephone

A G E N D A

P R O C E E D I N G S

10:00 a.m.

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

We have with us today -- well, we don't. We have committee members Harold Ray, John Stetkar, Mike Corradini and Joy Rempe. We might have others joining us as the day goes on. There's another meeting that's drawing some of us there.

Dr. Tom Kress is here as our consultant. Good morning, Tom. Mr. Mark Banks is the -- of the ACRS staff is the designated federal official for this meeting.

The purpose of today's meeting is to receive a briefing from Idaho National Laboratory and the NRC staff on the NGNP project. Department of Energy, the official sponsor of NGNP is here too.

During the subcommittee meeting on January 17 we received an update from INL on the TRISO coded fuel research. INL also briefed us about the work they've been doing with the NRC staff on development of a licensing framework for NGNP.

Today we expect to hear from the NRC staff regarding their review of INL work on the licensing framework development in addition to hearing from INL this morning.

Member Mike Corradini, Joy Rempe, Harold Ray and Dana powers have some potential conflict of interest. Hence they may be limited in discussion regarding their own work.

The rules for participation in today's meeting were announced in the Federal Register on March 25 of 2013 for an open meeting. This meeting is open to the public.

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

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

We will now proceed to the meeting and I call upon Mr. Carl Sink of the U.S. Department of

Energy to begin his introduction of the NGNP project. Carl?

MR. SINK: Good morning. Thank you very much for having us here again today.

As an introduction I'd like to just briefly review how we got to where we are today. Starting back in 2008 the Department and the NRC jointly issued a NRC-DOE Licensing Strategy as called for in the Energy Policy Act of 2005.

As part of that it identified that there were four key licensing technical policy and programmatic issues that may need Commission resolution before moving forward with a licensing framework. These included the acceptable basis for mechanistic source term calculation, the approach for using the frequency and consequence curve for selecting licensing basis events, the allowable dose consequences for those events, and requirements and criteria for using a functional containment that was anticipated for the NGNP.

We have continued this work and after review by the Nuclear Energy Advisory Committee in 2011 which recommended that we continue working with the NRC to develop a licensing framework Secretary Chu endorsed that with his letter forwarding the NEAC's report to Congress.

DOE appreciates the strong level of interaction that we've had with the NRC staff. We published, as I said before, the licensing strategy jointly. There was review and feedback on a large number of white papers covering various topics related to the NGNP. We've had about 18 public meetings over the last 3 years which were hosted by the NRC. Review of the NGNP responses to about 450 requests for additional information and feedback on that.

And then review of technology development plans specific to the NGNP project as well as approval of the NGNP quality assurance program description. And as we'll hear later on today they have given us feedback on the highest priority licensing issues that were described in a letter we sent them on July 6.

In that letter we specifically requested that those four key areas be highlighted again, licensing basis event selection, establishing mechanistic source terms, the functional containment performance requirements, and development of emergency planning, emergency planning zone distances.

So DOE has been focused and continues to be focused on resolution of key licensability issues to enable those applicants and the commercial sector to move forward under using a licensing framework. This framework which NGNP has proposed provides a process for assuring, along with the fuel qualification results which we talked about some last time, may touch again on today, that there is adequate protection for the public over a wide spectrum of internal and external events.

And we look forward to today's follow-on meeting following up on a couple of topics that where additional information was requested in our January meeting and also hearing the input from the NRC staff.

CHAIRMAN BLEY: Thank you.

MR. SINK: Thank you.

MR. SILADY: Good morning. My name is Fred Silady and my task here is to provide a summary --

MEMBER CORRADINI: Can I ask before we lose Carl, just so I understand. So, I guess I want to understand the path forward. I'm sorry. So, with the position paper -- or I don't know the proper terminology for what the staff has produced. That will then be used in what -- what's the next steps from DOE side given that the staff has responded some things with agreement, some with not so agreement, some with policy issues to Commission. What is the

DOE going to do in response to that?

MR. SINK: From our reading of the actual papers that were presented to us there was a significant level of agreement with the positions we had put forward. And we need to make sure that there is clarity in that there's nothing left vague with those.

And our understanding is that the NRC staff based on the feedback they get from the ACRS may make modifications to those position papers and re-publish them later this year.

MEMBER CORRADINI: Okay. And then? I'm trying to figure out is that going to be the end of it from NRC's interaction with DOE on the NGNP at this point in time and everything will be put on hiatus? I mean I'll ask the staff the same question. I'm trying to understand from your perspective where is it going.

MR. SINK: At this point in time then that will be the end of our -- on this particular topic. The interaction on this licensing framework development. We are still going to be continuing to R&D and interaction with the NRC and the staff on the fuels development, on the modeling, the work we're doing on HTTF. Other topics that we're doing jointly with the NRC staff will continued.

But so far as these meetings on this framework it was jointly agreed that we've come about as far as we can come with these topics for right now until an applicant comes forward.

MEMBER CORRADINI: Okay, so, all right. So you actually got to the point I guess I want to understand which is until somebody applies with a specific design both NRC will stand -- staff will stand down in discussions with you and then this effort on the DOE side will essentially cease.

MR. SINK: That's my understanding, yes.

MEMBER CORRADINI: On the licensing framework.

MR. SINK: The licensing framework, right.

MEMBER CORRADINI: Okay. All right. Then let me ask one follow-up question. Back in some year, I don't know what year, a few years ago the Commission specifically pointed to the NGNP as an example of exercising the technology-neutral framework process. This is kind of more of a question for the staff but since you're there. Has -- and you've seen, I think it's 1860? 1260? I can't remember. Has this been exercised from your perspective? In other words, has the technology-neutral framework process been exercised with this as an example to the point that it actually has been advanced. Any progress?

MR. SINK: I think so far as NGNP specifically and the high-temperature gas reactor project is concerned it's not completed yet. But we are anticipating moving forward with other advanced reactor concepts, licensing framework efforts with the NRC staff. Discussions about that are standing up right now for how that would move forward on additional projects.

MEMBER CORRADINI: Can you remind me what those are?

MR. SINK: Such topics as a fluoride-cooled high-temperature reactor, fast reactor, sodium-cooled fast reactor. Topics such as that.

MEMBER CORRADINI: And your intention, the DOE's intention is to use an approach similar to what you'd use for NGNP since this has shown some -- or this has been shown to be amenable with staff.

MR. SINK: Definitely. A lot of the work that we've done already in these areas is going to roll into that effort.

MEMBER CORRADINI: All right. Thank you. Sorry.

CHAIRMAN BLEY: Let me fire that up just

a little bit. I think I understood everything you said but is there a series of technical white papers and assessments of those papers -- what are your expectations? I mean we don't have an applicant, we don't have a design. Is there any agreement from your point of view that issues are settled at this point or how do you look on these white papers and the assessments of them by staff?

MR. SINK: I think that based on the feedback that we hear today and any additional feedback and questioning from the ACRS there's potential that these -- the four key issues could be settled.

CHAIRMAN BLEY: Okay, thank you.

MEMBER RAY: Well, Dennis, I think that that -- I had a similar question which is we've used the word "framework" here a great deal. I'm not sure we all know what that means which is implicit in the question you asked.

It really goes not I think to the good faith effort to provide responses but to how qualified the responses need to be, you know, how hedged given the limited information is available at this point in time.

So I think we shouldn't delay things here now but I do think a better understanding of what is meant by the Commission itself in terms of a framework being established, what does that really translate into is what I'm searching for here. I mean we can all do our best effort to reflect an effort but it's limited in its --

MR. KINSEY: This is Jim Kinsey from the INL. Just maybe to offer a point of clarification that might help a little bit.

The other piece of the licensing strategy that Carl mentioned that pointed out the four primary issues was a conclusion by both DOE and the NRC staff members involved with that working group that the NGNP could be licensed by adapting for the most part existing light water reactor regulations.

So when we talk about this framework we're really talking about the processes that would be used to work through that adaptation process. So if that helps to clarify. And these four items are key cornerstones or key foundations of that adaptation.

MEMBER CORRADINI: So here's what my concern is. Maybe I'm the only one that's concerned. But everything we had, there's so many issues that pop up and then rise up and then fall, and rise up and fall. I'm worried that when this falls it'll fall apart.

So I want to make sure it's very clearly documented where there are things that are -- I won't use the word "settled," I guess nothing is completely settled, but where it looks apparently things are in concert and where they're not in concert, and what are the action items to move forward.

And to me that's very important. That's why I was asking the questions of Carl was that -- is that if things aren't settled and there's things that need to be further done in fuels like there was some discussion about length of time and temperature and such things, then that's clearly identified as something that needs to be worked on. And there are certain things that are -- people seem to be okay with. I got the impression that the way in which licensing basis events were identified or at least the process by which they are staff seemed comfortable with. Those are clearly done so that when this gets dropped or put on hiatus it doesn't have to reinvented 6 months later, 6 years later, whatever it is.

MR. SINK: Yes. That's our intent as well.

MEMBER CORRADINI: Okay. And then my second concern is, and I'm sure the Commission did this, I'm just not sure -- and I'm wanting to ask the staff this. I want to make sure the staff really has responded to the Commission's directive for 1860 which is this is an example case for technology-neutral framework application and what are their lessons learned so that when you do come up with whatever the things you said you're going to come up with next we don't start from scratch.

We actually can say, okay, we did this with the NGNP, it's on hiatus but the same process or at least we can pick up here so we don't have to start from a, I don't want to say ground zero, but less than an optimal point.

CHAIRMAN BLEY: Now I think we're ready.

MR. SILADY: Good. Let's go to the first slide. At our January 17th session with you we went into some detail on these five areas. And we have a couple of more items to talk about at this meeting this morning before the staff as before. And we just wanted a summary, a very brief summary of the summary if you will of the January 17th meeting.

And these areas are the safety approach and design basis, the licensing basis event selection process, the mechanistic source terms, the functional containment, the siting source terms and of course fuel qualification and radionuclide retention. So I'm going to briefly go through those in a number of short slides here. Next page.

One slide for the safety approach and design basis summary. The top objective is not to disturb the normal day-to-day activities of the public outside the exclusionary boundary. So our quantitative requirement which is a design target of the project is to meet the protective action guidelines at an exclusion area boundary of roughly 400 meters. And that's for a very wide spectrum of events within and beyond the design basis.

We believe our safety approach and design basis summary is responsive to the advanced reactor policy. You can go back to the January 17 meeting for a tick by tick check-off of how we're responsive to it.

You obviously recall that we have a defense-in-depth system of barriers. They are -- many of them, they're concentric or one is completely inside of the other. At the level of the few elements, the helium pressure boundary and the reactor building they're independent. And these barriers collectively comprise the functional containment.

The emphasis has always been since the `85 time frame after Three Mile Island when we shifted from large HTGRs down to the modular HTGR where we deterministically said hey, we're going to size and we're going to configure the reactor in a long, slender, annular core. It's been on retention within the -- at the source within the radionuclide particles and within the fuel element.

So to do that we need to do the following three sub-functions which we talked about, the passive heat removal, the control of heat generation and the control of chemical attack. Next page.

The licensing basis event selection summary is that we're going to determine when top-level regulatory criteria must be met. The top-level regulatory criteria are the quantitative direct -- top-level regulatory criteria are -- have three things. We went through all the regulations, NRC, EPA and so on and we screened them so that we could figure out what we needed to design to.

And if they're quantitative so that you can design to them, they're direct measures of consequence, of risk to the public, and they're technology-neutral, they're generic. So we select then during the design and licensing process with the risk insights of a full-scope PRA that considers uncertainties we select these events.

And they fall into categories. Those that you expect during the life of the plant, those that you don't expect in a plant lifetime but which might occur if you had a fleet of plants, like several hundred, and those that aren't expected even in a large fleet of plants, the beyond design basis events, the events that you don't design for with conservative margins but which you have the capability to respond to and still meet our top requirements.

And there's a fourth category. This is the traditional Chapter 15 events. And we derive those from the design basis events by assuming only safety-related SSCs respond successfully.

So the DBEs and the AEs and the beyond design basis events, those we put on a frequency plot which is the next plot. And those have the entire plant responding. So we see the interplay between what is safety-related, not safety-related, and so on.

MEMBER CORRADINI: So, just to clarify.

MR. SILADY: Sure.

MEMBER CORRADINI: So for the DBAs, all the similar assumptions we're familiar with for light water reactors apply.

MR. SILADY: I wouldn't claim to know all those assumptions. But we --

CHAIRMAN BLEY: -- assume a single failure. The failure criterion is the one --

MR. SILADY: If the event had single failures in it, fine. If it had multiple failures in it, fine. That we get from the DBEs.

Then we just look at it with the safety-related SSC. So I think the answer is no with regards to that. Not that we wouldn't pick that up in the beyond design basis events, perhaps, but we take multiple failures within the design basis event region. And we're looking at more than one reactor as you'll see.

MEMBER STETKAR: I wasn't here in January so perhaps you covered some of this, but as I read through it I had some confusion about the notion that you just described. You said that the DBAs are derived from the DBEs, assuming that only safety-related systems are available. Is that correct?

MR. SILADY: Yes, that is correct.

MEMBER STETKAR: Okay. Suppose I have a beyond design basis event scenario whose frequency is in the beyond design basis event area. Because non-safety related systems have been included in that model. So for example, the frequency is, pick a number. One --

MR. SILADY: 10^{-6}.

MEMBER STETKAR: 10^{-6} and of that 10^{-6} 10^{-3} of that is because of non-safety related systems. If I didn't take credit for those systems the frequency would be 10^{-3}. Is that scenario then a candidate for a design basis accident? Because it's not a design basis event in the general concept.

MR. SILADY: We're going off a little bit but I think it's a very crucial point so let me just try to summarize it succinctly.

MEMBER STETKAR: I'm not sure we're going off because I want to understand what a design basis accident is.

MR. SILADY: Right. But we talked about this in January. There are beyond design basis events that have high consequences that would not be able to meet the dose criteria for the design basis region, the 10 C.F.R. 50.34. Just as we want to make sure that the DBAs that mitigate and stay within the design basis region, 10 C.F.R. 50.34, we want to make sure that beyond design basis events with high consequences don't float up. And so we make things safety-related to have the requisite reliability to prevent those.

So that is how beyond design basis events can have requirements put on SSCs that then become safety-related. We had a nice little chart that had on the FC chart we had one area where we're mitigating DBAs and one area where we're preventing beyond design basis events.

So, because if the beyond design basis event doesn't have a consequence that's going to exceed 10 C.F.R. 50.34 what the fraction is of the plant that responded that was safety-related and what the fraction was non-safety related isn't as material as it is if it would violate 10 C.F.R. 50.34.

MEMBER STETKAR: I'll have to go back and read the transcript.

MEMBER CORRADINI: You're going to show that curve I assume.

MR. SILADY: We're going to show the curve but we're not going to -- we can pull up the backup if we need I believe.

MR. KINSEY: I think the backup will be a good idea.

MR. SILADY: Yes.

DR. KRESS: Let me ask you a question about that. When you look at an SSC in one of these events and look at the uncertainties associated with it to see if it might move you into another frequency or another consequence. You just look at one SSC?

MR. SILADY: We look at the sequence and we look at the function. And then in that sequence there are certain SSCs that are available and not available. And we look then at if the event straddles the design basis region let's say and the beyond design basis event we say well, our certainty is not sufficient to say it's in this region or in that region. We'll look at the consequences of it against both as if it first were in the design basis and secondly it's in the beyond design basis.

DR. KRESS: If it takes, say, two or more or three SSCs to move you in one direction or the other beyond this top-level criteria would all three of those or however many it took --

MR. SILADY: That would be a design choice as to one or more of them need something to be tightened up in order to meet the requirements.

DR. KRESS: And what criteria do you use for that choice?

MR. SILADY: Making sure that the event meets the dose criteria in the design basis region or the QHOs in the beyond design basis event region.

DR. KRESS: Yes, but suppose further or the probability of one failure, say three SSCs lose you and the two won't. Would you make all three of them safety-related or?

MR. SILADY: If -- you cannot make one system of a high consequence beyond design basis event keep you in that low-frequency range.

DR. KRESS: You're looking at one at a time SSC.

MR. SILADY: If you cannot make one -- that would be the designer's preference, to put the reliability into that. Then you go to a second one, obviously.

DR. KRESS: But if it takes two of them to move you would you make both of those SSCs?

MR. SILADY: Yes.

DR. KRESS: You know one of them won't do it by itself.

MR. SILADY: That's right.

DR. KRESS: So you'd make two.

MR. SILADY: Yes.

DR. KRESS: Well, how about four of them?

MR. SILADY: I don't think for a given function we have four heat removal systems.

DR. KRESS: You probably don't for a given function but you may have three.

MR. SILADY: Yes, yes. And generally we

find by using passive SSCs like the reactor cavity cooling system we can make it sufficiently reliable that we can say that one alone is sufficient.

In other functions such as control of heat generation we may need two reactivity systems. We find that generally we do for some rare events.

DR. KRESS: So you would make both of those.

MR. SILADY: Yes.

DR. KRESS: Even though one of them won't do it by itself.

MR. SILADY: They work in tandem to get you in the successful -- for some sequences only one's required. For other sequences maybe both are required. It's not that black and white. You have to look at each of the licensing basis events.

MEMBER REMPE: Fred, remind me because I've forgotten. Is the RCCS safety-related?

MR. SILADY: It is. For heat removal, yes. All right, well this is good. We didn't expect maybe that we'd be going back to January 17 but I'll be happy to answer the questions.

MEMBER CORRADINI: We still haven't moved off of the 17th. So let me repeat the question that Dennis and John asked just so I'm clear. So if it is safety-related and it is going to be used as part of the analysis for the DBA, the single failure criterion applies.

MR. SILADY: I don't think we've said that. We're always after getting the reliability. And we don't think in some cases single failure is sufficient. And that diversity may be better than redundancy.

MEMBER CORRADINI: So can I say your answer back to me a different way?

MR. SILADY: Yes.

MEMBER CORRADINI: If I have system X and I have one of system X, and I have system Y and I have one of system Y, your redundancy is that if X fails Y is there to perform the function.

MR. SILADY: If needed to stay in that region and meet that requirement.

MEMBER CORRADINI: So therefore X and Y must be safety-related.

MR. SILADY: Yes. Yes. We have that, let's say in the case of the control of heat generation. We have a control rod system and it has a lot of redundancy in it, banks of different rods and so on and in the I&C in the protection system there's redundancy. But parts of it are only single.

And then we have a reserve shutdown system which is completely independent that also drops by gravity and so on. And it has different numbers of hoppers so it has redundancy. We didn't go in though and say thou shalt have redundancy and take the single-failure criteria. Usually those things are the first thing you do in terms of reliability. And they're built in for investment protection reasons and so on. So you have a full plant design that's looked at all your requirements for normal operation, investment protection, availability as well as safety.

I think we'd be hard pressed to say that we found any sequences that violate the single failure criteria that we're not doing anything about. I mean that's -- we're --

MEMBER CORRADINI: Well, that's kind of what I'm getting to.

MR. SILADY: Yes. But to take it as a rule, thou shalt use the single failure criteria, is kind of why I'm pushing back a little bit.

MEMBER CORRADINI: Okay.

MR. SILADY: Okay. So we've gotten off a little bit into the fourth bullet. And we -- the safety classification system focuses on mitigation for a spectrum of DBAs to successfully perform required safety functions. But we were trying to keep it succinct and so we didn't add also to prevent high-consequence beyond design basis events which could have been on the slide as well. And I think I've summarized that. Let's go to the next page.

Now this is the top-level regulatory criteria which are in solid blue that we took out of the regulations. Placed on a frequency axis. And in some cases it's easier to do than in others.

And so this was the proposal that we've made. It was actually made in the eighties with the DOE-sponsored MHTGR program. And it has evolved a little bit in terms of terminology and so on. But basically there's a region that is the anticipated. And we decided to take it down to once in 100 years. There's another region that goes between the lower level of the anticipated event region. It goes down to the 10^{-4} and so on.

Well, what are those numbers? What are those frequencies? They're the frequency of an entire event sequence, not just the initiating event. And they are for an entire plant of multiple modules.

DR. KRESS: Let me ask the question.

MR. SILADY: Yes.

DR. KRESS: To me that means, of course,

in order to see where things fit in that thing you have to specify ahead of time how many modules you're going to have.

MR. SILADY: Yes. You have a design.

DR. KRESS: You have a design, in other words.

MR. SILADY: Yes.

DR. KRESS: And you have so many modules. The tendency in my mind would be then, I'd want to select the number of modules so that I get just about as close as I want to to that top-level regulatory criteria without exceeding. Do you have a criteria on how close you're going to let it get by the number of modules?

MR. SILADY: We really don't do it the way you say. I mean we look at the other stakeholders in terms of the users and the operators and our requirements coming from them on how many modules make it economic, make the O&M optimum that provide the demand for the electricity, the steam nearby.

DR. KRESS: Even though you may end up pretty close to that line. Do you have a criteria for how close you're going to let it get depending on the uncertainties associated with the --

MR. SILADY: We don't have a quantity

requirement at this stage prior to a full conceptual design. But in past projects we've seen like the MHTGR that were orders of magnitude anywhere to the left of the solid line and even the dashed line which of course as you can see is the design target which is more stringent. So we haven't had to say thou shalt be a factor of 20 or a factor of 5 or whatever.

And that tradeoff with modules -- I do have a backup that shows the MHTGR cases. Can we put that up at this point?

CHAIRMAN BLEY: Slide 30?

MR. SILADY: Yes. This was presented.

MEMBER CORRADINI: I remember that --

MR. SILADY: Yes.

MEMBER CORRADINI: I think we did see this one, yes.

MR. SILADY: And we had already started to talk a little bit about design basis accidents as well as design basis events and how the design basis accidents only have safety-related SSCs. Originally this plot was just of the first three categories of LBEs. They were called anticipated operational occurrences. They were called design basis events as they are now. And they were called emergency planning basis events back then where now we're calling them beyond design basis events.

DR. KRESS: How many modules does it --

MR. SILADY: And it had four modules.

DR. KRESS: Four modules, okay.

MR. SILADY: And there have been HTGR designs that have had up to 10. But no more that I'm aware of. And so we saw where those events lied on the diagram. And we're using this for design as well as licensing. So the ones on the abscissa that have zero dose were very important to us because there was something keeping them inside the acceptable line that if it didn't work it might fly over to the right-hand side here. So we had to know which ones those were and specify their capability and reliability and so on.

The point with the green on this chart is that these DBAs when you deterministically start assuming from your DBEs that you don't have systems they're lower frequency. There are some over there on the abscissa that are pretty close to the same frequency and with the uncertainty bands they were DBEs.

But there were quite a few DBEs on the high-risk, the high-consequence events that were in the mud if you will, that were below that needed to meet the NRC safety goals, below 5 times 10^{-7}. And there are some that don't even show up on this plot that goes down to 10^{-8}.

So the DBAs are not -- don't come out of the frequency consequence plot. The DBEs do and then you use your deterministic approach that we use in Chapter 15 to make sure that the safety-related alone could help you meet 10 C.F.R. 50.34.

So these events that are way low that normally would be compared to the safety goals, the QHOs, the Q fatality safety goal there only have to meet the 10 C.F.R. 50.34 and your Chapter 15.

All right, thanks for that diversion. Are we good now maybe? Let's go back to my other plot and see if there are any points there that -- just wrap up with.

I wanted to make sure that you understood that it's the event sequence, that it is per plant year and it's going to have a full-scope PRA so it's not just going to be reactor, it's going to be other sources like spent fuel and so on. So you might have spent fuel shared or you might have it per reactor module. All that's in the mix here.

DR. KRESS: So you're going to specify up front though how many modules.

MR. SILADY: Yes, yes.

DR. KRESS: You would let the person buying the plant say we want five modules for our side, and then you'll put --

MR. SILADY: I can't predict how the business arrangements might be. But we will have a design that probably mock 1 is four reactor modules. And then somebody comes in and say oh, I only want one now. And so then we'll say, okay, well we have to -- will you have the interest in someday having four and they may say yes.

And so we'll build the shared things for the four and we will sequentially add them. But the PRA and the selection of the LBEs has to consider the four because someday you're going to build that out perhaps.

DR. KRESS: Well the way you do that is just find the consequences for the one module and just --

MR. SILADY: That's not --

DR. KRESS: -- multiply by the number of modules.

MR. SILADY: No, no, no.

DR. KRESS: No, you can't do it?

MR. SILADY: You can't do that because

you're going to have some events that affect all four of them. That's the beauty of the approach.

DR. KRESS: How do you treat those?

MR. SILADY: You say, okay, I've got a seismic event, or I've got --

DR. KRESS: Station blackout.

MR. SILADY: -- station blackout and that's the initiating event. And then you look at what the chances are of it affecting one or more modules. And you work your way across the event tree.

DR. KRESS: Well, let's take seismic.

MR. SILADY: Yes.

DR. KRESS: You have to say yes, it's going to affect all four modules.

MR. SILADY: Right, but it may take out let's say the main heat transport system for all four. But it may not take out all the shutdown cooling systems for four.

MEMBER CORRADINI: How do you know -- I can almost guess Tom's next question is how do you know with any certainty that it isn't anything but a multiplicative on the one module effect. Is that where --

DR. KRESS: That's where I'm -- you could read my mind. But my conclusion was that since there's so many SSCs that are relatively independent of the initiating event that probably the way you treat the initiating events that affect all of the modules at the same time, you probably go ahead and calculate the consequences due to one module and just multiply the number of frequencies.

And just like you would do the internal events. It's the way I think you'd probably treat that.

MR. SILADY: If it was the example loss of offsite power. We have a lot of passive systems.

DR. KRESS: Yes, that's what I had in mind.

MR. SILADY: Yes. And so it might be bimodal in that case in the sense that all the passive systems had an independent failure because the same maintenance crew did the maintenance on all of them or the same manufacturer or whatever. Low-frequency albeit. But you could think of things that can get passive failures as well. And then you've got the loss of offsite power and the loss of onsite power and so on. And you end up with some failures in the low frequency range.

But there are shades. And we have found that there are events where it's like 30 or 40 percent of them are all reactors are taken out. And 20 or 30 percent are just one.

DR. KRESS: So the consequences would be affected --

MR. SILADY: Yes.

DR. KRESS: -- rather than just a single --

MR. SILADY: Yes.

CHAIRMAN BLEY: I'm going to interrupt because we're repeating I think a lot of what we talked about last time. And you have some new stuff you're going to get to.

MR. SILADY: I want to.

CHAIRMAN BLEY: And all of you do. So I think we're going to have to reduce the amount of revisiting the old material and look for those responses aimed at -- probably that were aimed at questions --

MEMBER STETKAR: Can I ask one quick one?

CHAIRMAN BLEY: Sure.

MEMBER STETKAR: And you can turn me off if this was discussed in January also. I read quite a bit of the study so I don't want to hear high-level things, I want to hear details.

I understand how you're treating

uncertainty on the Y axis. I understand how you're doing that.

I do not understand how you're treating uncertainty on the X axis. In particular, you state that in the AOO region you will compare only the mean value of the consequence with the high-level goal. In the DBE region, whatever that means, you said you're going to take the upper bound of the mean which I'm assuming you really mean the 95th percentile of the uncertainty because I don't know what the upper bound of the mean means. You're going to compare that with your goal.

In the beyond design basis event you will compare only the mean with the goal. I don't understand that rationale. Because if I take a beyond design basis event with an uncertainty bound in the X axis and I move it up then I don't know what I'm comparing on the consequence scale. And I'll just leave it there because maybe you discussed it in January. So I'd like to understand how you're treating those uncertainties in those ranges because all the high-level stuff you say sounds good until you look at how it's going to be done in practice.

MR. SILADY: For all the events.

MEMBER STETKAR: For all of the events.

MR. SILADY: For all the events, the AEs, the DBEs, the beyond design basis events we do a consequence uncertainty.

MEMBER STETKAR: Yes, you do.

MR. SILADY: And so we've got the full distribution everywhere.

MEMBER STETKAR: Yes, you do, and I'm not arguing with that. I'm saying that you're picking and choosing what parameters of that uncertainty distribution you want to compare with your goals.

MR. SILADY: We tried to follow industry practice in terms of --

MEMBER STETKAR: Picking and choosing what parameters of the uncertainty you're choosing with each area, period.

MR. SILADY: We are, yes.

MEMBER STETKAR: Okay. I don't know the rationale behind that.

MR. SILADY: Okay. It is discussed in the white papers.

MEMBER STETKAR: I read it and I didn't understand it. It just says other people have done it in the past.

MR. SILADY: Okay. I'm being told that maybe in a couple of slides.

MEMBER STETKAR: Okay, if you're going to address it then we'll get to it.

DR. KRESS: One other issue I have with -- just popped out of a regulatory criteria being stair step has bothered me all along.

MR. SILADY: Yes.

DR. KRESS: And I would have made straight line --

MR. SILADY: ISO risk. Or maybe it was an adverse --

DR. KRESS: Well, I mean it's maybe non-risk averse. But I would have made it a straight line. That way I know exactly how far away I am from the boundaries. But if you're pretty close with this stair step I'm never quite sure how close I am to the boundaries and how to -- because there's some arbitrariness to these stair step methods in terms of the slope.

MR. SILADY: There is in more than -- in probably one case for sure.

DR. KRESS: Yes. So I would have elected to have a little conservative straight line on the slow glove plot for my top-level criteria. Did you consider that?

MR. SILADY: We took these from the

regulations and the QHO ends up looking like a straight line. And so you're happy with it.

DR. KRESS: yes.

MR. SILADY: The 10 C.F.R. 50.34, we had it all of 25 rem and back in the eighties we said -- the staff gave us feedback, oh, for a higher frequency DBEs we wouldn't want to come that close. Take 10 percent of it. So we made that a slope based on that input.

The 10 C.F.R. 20, we believe we have it exactly the way the regulation says and we can't change the regulation. It's summed over all the events and you can't exceed that. So depending upon what the frequency is you can have an event that occurs twice a year so you only take half of it.

So it is what it is. We would love to have something come to us that says here's the NRC frequency consequence curve and here's the basis for each kind of like they did for the safety goals. Office of Policy Evaluation could come up with.

DR. KRESS: Well, you could have drawn a line between the bottom points on those stair steps and make a straight line through it. And it would be a little bit conservative but --

MR. SILADY: That's true.

DR. KRESS: -- it would take away all the -- any arbitrariness.

MR. SILADY: You're talking about going from 10 C.F.R. 50.34 out to 10 C.F.R. 20.

DR. KRESS: Yes.

MR. SILADY: This was a starting point that where it's not about the specifics, it's about the process. There are bigger questions than where the line is.

DR. KRESS: Yes. Well, you know, it depends on how close you get to that line as to how you're selecting SSCs as to whether you cross over or not. It would make a difference if you had a straight line as opposed to the stair step I think as to which ones might cross you over into the unacceptable region. That was one of my issues. I don't know, I just right now don't know how to deal with that.

MR. SILADY: Okay.

DR. KRESS: One way you can deal with it is be sure you don't get very close to the stair step.

MR. SILADY: That in essence is what we have found to this date, yes. Let me go on.

MEMBER REMPE: Actually, I have one quick clarification too. You're saying --

MR. SILADY: Dennis, I'm really trying to go on here.

MEMBER REMPE: I know. If you have 10 versus 4 versus 2 modules you're almost implying that you might have different events, you might have different systems you designate as safety-related or you might move the boundary.

MR. SILADY: No. No.

MEMBER REMPE: What will you do?

MR. SILADY: A vendor in all likelihood is going to have a four-module design or X-module design. He's going to offer that.

MEMBER REMPE: And I know of other vendors who've come in and said even though we're way over the safety limits we're not changing the design because it's too expensive. And so what the design is is what it is, and they just delete the cost because it's too expensive to make modifications is why I'm asking the question.

MR. SILADY: Well, where I was going is as the market develops you might come out with a second package. Instead of four you see there's a need for a two-pack, or you see there's a need for a six-pack. And so you offer then a selection. Do you want the two or the four, you know, whichever two you select.

CHAIRMAN BLEY: I think Joy's point is the one that's been bothering me. You go down, it's no big deal I think. You go up to more units on the site you may have to really change the design and that seems, you kind of said that, it's a pretty unlikely thing to see happen.

MR. SILADY: Yes, yes. You --

CHAIRMAN BLEY: So from early on --

MR. SILADY: Early on --

CHAIRMAN BLEY: -- somebody's got to decide how you're going to design this thing.

MR. SILADY: Yes. Exactly.

CHAIRMAN BLEY: For the maximum number of --

MR. SILADY: Yes, that's right. And there's all these questions about what if you don't put it on greenfield and you put it where it has existing reactors. Is there any budget, is there any room for you to put yours on there and still overall site-wise meet the requirements. But we're not going there today, okay?

CHAIRMAN BLEY: Okay.

MR. SILADY: Next page. I did it.

(Laughter.)

MR. SILADY: We'll go back. Okay. Later. The frequency took a lot of discussion but it's pretty straightforward. You're going to use the same frequency in all cases, event sequence, mean and plant instead of a reactor. But now we're inheriting all these different requirements out of the regulations and some of them are at the EAB, some are at the LPZ, some are, you know, you can read the chart. And we talked about this in the earlier meeting.

And so just for ease of presentation we're plotting everything at the EAB. And almost all the regulations now have gone to the total effective dose equivalent.

So when it came time to do NRC safety goals even though we're plotting and showing where the points are relative to the -- at the EAB we would do it per the regulation. You know, the acute to -- as shown there to 1 mile, the latent at 10 miles. And it would be the complementary cumulative distribution function on all the accident rule set that that comes with that particular requirement.

DR. KRESS: If you meet your EAB criteria you're almost sure to meet those other two.

MR. SILADY: That's right. And that's why we did it this way.

DR. KRESS: Yes, but there's one missing there.

MR. SILADY: Okay.

DR. KRESS: That's the societal risk. The total cost of an accident. This goes up forever, or maybe 50 miles. But I'm not so sure you automatically meet that when you meet the EABs. But I don't see anywhere where you're looking at that. The PRA preliminarily looked at the total cost of an accident. And taking all the events including beyond design basis and everything. But I don't know where that fits into your system.

MR. SILADY: Well, if it comes from the NRC or --

DR. KRESS: There's no requirement yet from the NRC.

MR. SILADY: I know, that's the point. We screened the current regulations. But we have from our user, in the MHTGR days we had an investment protection. And it had an FC chart.

DR. KRESS: Oh, you have one.

MR. SILADY: And then the C --

DR. KRESS: It has dollars for the C?

MR. SILADY: Yes, exactly right.

DR. KRESS: Oh, wonderful. I love that.

MR. SILADY: We're off into another topic.

DR. KRESS: Okay.

MR. SILADY: Okay. At 10^{-3} you couldn't be down for 6 months. At 10^{-5} you couldn't have plant write-off, et cetera.

DR. KRESS: Somewhere I'd like to see that eventually.

MR. SILADY: Okay. I can send it. Next page, please.

Now, we've touched on this already. We've got different requirements shown in the green across. We've got these categories in the column, first column. They come to us with different accident rule sets.

And so although we're doing our uncertainty distributions for all the top three categories we pick whatever value we need for whatever the requirement is. And our understanding is on this page, that for the 10 C.F.R. 20 we cume the events and we look at it at the EAB for that 100 mrem.

And when it comes time for the DBEs or the 10 C.F.R. 50.34 it's just the DBEs and we look at it upper bound, 95 percent. For the emergency planning you want to know what the real expected, the mean values are going to be to compare to your PAGs, whether you have to move or shelter people.

For the QHOs we know that that's pretty

well spelled out. So once we've derived there's a space between the top three which can go on an FC chart very neatly and the DBAs that are scattered everywhere.

We use the upper bound against 10 C.F.R. 50.34 again at the EAB even though there's an LPZ. And we do this typically for 30 days but we will look at 2 hours of course, the worst 2 hours.

And all the while you have to keep in mind that our design objective, what we're trying to do is have that EPZ be at the EAB. Next page, please.

MR. KINSEY: Excuse me, Fred. Before you move on. Does this help to answer the question about where we do consequence uncertainty and where the limits come from? And the method --

MEMBER STETKAR: It doesn't help to answer where the limits come from except that you feel that your interpretation is somebody else has told you to do it this way. So I'm going to ask the other people that you feel have told you to do it this way.

MR. SILADY: Well, it's not just telling us but we've looked at a lot of different sources, regulatory and submittals and so on, and this is what we believe is current.

MEMBER STETKAR: I'll ask the staff.

MR. SILADY: Okay.

MEMBER STETKAR: See if they agree.

MR. SILADY: There is some disagreement on this as well, so this is a good thing to bring to mind.

MEMBER STETKAR: I didn't quite see that disagreement. I'll ask the staff.

MR. SILADY: Okay, very good.

MEMBER STETKAR: Keep you going.

MR. SILADY: Next page.

MEMBER CORRADINI: So your objective -- I just want to get to your objective.

MR. SILADY: Yes.

MEMBER CORRADINI: Your objective is to make the EAB the EPZ.

MR. SILADY: Correct.

MEMBER CORRADINI: And if necessary you'd have to grow the EAB to make that occur.

MR. SILADY: If necessary. But there's lots of other things we could do. We can change the design. We can do more research. We can sharpen the pencil. You do all the designer tricks in order to find a way to make your design such that you wouldn't have to shelter or evacuate anybody offsite.

MEMBER CORRADINI: Okay.

MR. SILADY: Next page. This is the discussion we've already had of how we get the design basis accidents from the DBEs. We don't get them out of the air. We get them from the DBEs. That gives them a firm, systematic, nothing's going to drop through the cracks because we're going to put all the focus on the PRA that gave us those DBEs.

The DBAs are not derived from the beyond design basis events. It's the events that are in that frequency range in the DBEs that go into Chapter 15 that we're going to look at deterministically. This is one of the major assumptions that blends the probabilistic with the deterministic. Enough said on that, let's go to the next page.

MEMBER STETKAR: No, not enough said. That last bullet is the thing that I'm not understanding. Because if I do a risk assessment and I look at I'll call them sequences. I look at a sequence and I have a bunch of things that fail and a bunch of things that succeed. And a bunch of things that fail get me down to 1 times 10^{-6}. And that's a BDBE.

MR. SILADY: Yes.

MEMBER STETKAR: And it has some consequence.

MR. SILADY: Yes.

MEMBER STETKAR: So I have uncertainty about that consequence. Now, if I look at the bunch of things that have failed and I say, well, half of that bunch is non-safety related. I'll call it non-safety related. Now, if I -- do I assume that that non-safety related stuff cannot work when I think about design basis accidents?

MR. SILADY: In our framework we think of design basis accidents in the DBE space. In the beyond design basis event space we only think of those that are high-consequence that would exceed 10 C.F.R. 50.34 that we would then need to make things safety-related to keep them low-frequency.

So the only time in which the DBA or the safety-related comes into play is either from the DBEs to mitigate them, to meet the consequences, or from the high-consequence BDBEs that are -- if we looked on the chart they're over to the right that we don't want to have rise up because there --

MEMBER STETKAR: But they're still not considered DBAs but they might have -- you might define safety-related equipment because of that criterion.

MR. SILADY: Yes. And actually --

MEMBER STETKAR: I got it.

MR. SILADY: Okay.

MEMBER STETKAR: Next slide.

MR. SILADY: Thank you. Good, progress. Next slide, please.

Now the functional containment is a topic that we covered on the 17th as well. Our upper tier -- and I've already told you that it's all of those barriers that are for lack of a better word concentric or nested, and that there's independence and so on between the helium pressure boundary and reactor building and all the fuel.

What we do here is we set intentionally that we're going to have requirements on retaining radionuclides within the fuel. We really want to put the focus on retention at the source.

But we're going to look at what the other barriers do for us as well in terms of helping us have additional margin to our requirements. So, the standard that we're looking for performance here is characterized by during normal operation retention within the fuel so that we have a really relatively low inventory within the helium pressure boundary.

Then if you have a leak in the helium pressure boundary it won't exceed the requirements offsite. That's the idea.

And then secondly we're going to limit releases throughout the spectrum for the off-normal events. Whether it be an early release or an early and a delayed and so on.

Finally, our functional containment system has the capability to control leakage when you think of it in the full five barriers. Next page.

Our summary of where we are, and this is largely from the MHTGR and the PBMR and some of the other designs that have been before in the pre-application interactions over the decades is that we can release that which comes out during normal operation from the helium pressure boundary and meet 10 C.F.R. 50.34.

Our limiting LBEs tend to be, the risk-significant ones, have an initial release from the helium pressure boundary. Because if you don't fail the helium pressure boundary you aren't getting anything out.

And there's a range of possibilities there. There's leak sizes and leak locations, and there's the possibility of the relief valve. And the high-risk events really are not the big breaks in the helium pressure boundary but it's that relief valve that goes off because the steam generator is at higher pressure. You can have a steam generator tube leak and that in certain sequences if the water keeps coming that relief valve lifts. So that's the initial release.

And then you have the larger delayed release from the fuel which takes as you know 2-3 hours because of low power density and high heat capacity.

The next two statements are significant. We will meet 10 C.F.R. 50.34 without consideration of reactor building retention. We will meet the PAGs with the entire functional containment including the reactor building.

As you're going to hear later in the day the reactor building safety-related primarily for structural reasons. Next page.

The summary on the functional containment mechanistic source terms is that there's a blend. We're using retention at the source and the intrinsic properties and the passive features in order to meet the requirements. It's consistent with the advanced reactor policy. It's consistent with discussions of the containment function of mechanistic source terms in various SECY documents and with the approaches that have previously been reviewed dating back to the eighties. Two or three different times as we've done with NGNP, we did it with the MHTGR, with PBMR Exelon and with PBMR alone. Next page.

Here's the last page before I turn it over to Dave to talk about the fuel area. The siting source term is essentially -- our approach to it is essentially the same as what we did in the eighties and which the staff reviewed in NUREG-1338.

It's consistent with the discussions of containment function and mechanistic source terms in more recent SECY documents. It implements a modular HTGR-appropriate interpretation of that footnote that started off in 10 C.F.R. 100 and which is now in 50.34 and 52.79 regarding siting.

Limiting DBAs are what we use to evaluate, to determine the SSTs. Remember our chart with the blue dots? We used the big ones that are way down in the mud as our -- source terms. Five times 10^{-7}, even some of them below that, but above 10^{-8}.

Back in the eighties the staff still wanted more what-ifs and deterministic flavor. So they said hey, go look at these other bounding event sequences is what they called them. And we had to look at cross-vessel failure, double-ended guillotine.

We had to look at all the rods being pulled and we had to look at -- and we did it. And we intend to do that again.

And the reason we did it is to assure there were no cliff edge effects. That's how we used those what-ifs, so that we know the safety terrain, the topography of the land, that there's nothing lurking out there both in the weeds.

Well, I've had fun. I hope I've covered the ground. I've used way too much time and I apologize but I've left a little bit for Dave I believe.

CHAIRMAN BLEY: Not really, but it's only our lunch, so.

MR. PETTI: So last time you heard a lot about the fuel program, what we're doing to support the design. So just in summary that we do have a large fuel program providing data under NRC-accepted QA program to really understand fuel performance and fission product behavior for laying the technical foundation needed to qualify the fuel made to fabrication process and product specifications within an envelope of operating in accident conditions that we think will bound modular HTGRs.

The results to date that I talked about

last time are consistent with the design assumptions about fuel performance and radionuclide retention that have been historically used by reactor designers. And we're getting data to support model development and validation.

And in one simple statement the results to date support the design basis that you heard about include the approach for functional containment and mechanistic source term.

In terms of what are our key results, we have a vastly improved understanding of TRISO fuel fabrication. We spent significant effort to improve our understanding over the historic German process. Much better fabrication and characterization by the fuel vendor largely because measurement science is just better than it was in 1978 and 1980. Our ability to do certain parts of the fabrication equipment is better. We're actually leveraging the computer industry and making chips, one of the key components in the fabrication of fuel is the same component.

We've had an outstanding irradiation performance. You have a large statistically significant population in TRISO fuel particles at high burnup and high temperature HTGR conditions. And we have confirmed the expected superior irradiation performance of uranium oxycarbide fuel at high burnup which has been a point in the fuel community for decades.

We're in the middle of wrapping up our post-irradiation examination of the first fuel AGR-1. And as many know silver tends to come out of this fuel. We got a lot of silver out of the fuel in AGR-1. Generally consistent with the model predictions to date but no cesium release from intact particles under irradiation. Any cesium we see in the fuel matrix after irradiation, it was a defective particle in that capsule.

There was no palladium attack or corrosion of silicon carbide despite large amounts of palladium outside the silicon carbide. Percent level of palladium went through the silicon carbide but the silicon carbide is fine as evidenced by the fact that there's no cesium outside the silicon carbide.

And we have done some safety testing. I think we just finished number 7. Hundreds of hours at 16-, 17- and in red there is 1,800. We just completed an 1,800 degree C test demonstrating the robustness of the fuel. And we'll talk about it in the next slide.

So the accident safety testing is well under way. We simulate this core conduction cooldown that you see in the plot. And what we do is simple, right now, isothermal testing for hundreds of hours at 16-, 17- and 1,800. We have done -- it says six tests. I was losing track -- have been completed.

We will actually try to do a test sometime this year, early next year that follows the purple line, the actual time/temperature results for the core to compare to the isothermal. And we are thinking about deconsolidating the particles and just heating up the particles and not the matrix because our releases look like it's just material that had diffused out into the matrix under irradiation and the particles were not releasing under these accident tests. We just want to be able to confirm that. But it does look like that that's the case.

MEMBER CORRADINI: So can I? I'm not a fuels person. I was hoping there would be somebody in the committee that knows the right question to ask but since -- so your last point, I guess I have a number of questions.

So your last point is that you think it got there from the fabrication event and during the irradiation testing, or it leaked out over long time spans during irradiation testing? That's what I wasn't clear on.

MR. PETTI: During irradiation depends on the fission product. Some fission products will diffuse through the silicon carbide.

MEMBER CORRADINI: The metallics.

MR. PETTI: Some of the carbides besides silver which is not safety-significant. But europium and very low levels of strontium we see in the matrix, 10^{-3} to 10^{-5} fractions, small levels.

Then we put a twin compact because that one we had to destroy to get that number. Then we take the one sitting right next to it and put it in the furnace and the release is about the same as when we dissolved it and said what's in the matrix.

The releases are flat over time. You put it in the furnace, you get to the temperature, they never increase which says it what sort of came out and then just slowly, slowly on a log plot.

MEMBER CORRADINI: You're just essentially cooking it out of the --

MR. PETTI: Cooking it out of the matrix, not out of the particles.

MEMBER CORRADINI: Okay.

MR. PETTI: So. So to absolutely confirm that it would be nice just to heat some particles and that's what I think we're planning to do.

MEMBER CORRADINI: All right. So the same question is the staff in reading their document were concerned about time and temperature. And are you going to address that somewhere in these slides? Because I want to get a feeling because I don't appreciate all of this.

MR. PETTI: That had to do more with I believe irradiation testing. I think the accident testing time and temperature is well in excess.

I showed a plot the last time that I was here that tried to capture because there's so many different transplants in the irradiation capsule. How much time, what fraction of the fuel spent at what temperature. And we've -- now that we have that all in place we'll do that for every capsule to show.

But for instance, AGR-1, 5 percent of the fuel spent time above 1400 degrees C for 100 days. So there is a lot of fuel that got very hot. So we're trying to capture that metric because it's a hard metric to capture.

CHAIRMAN BLEY: Your uncertainty calculations, and I've just started reading the reports on the temperature, thermocouple data analysis and the uncertainty. You came up with, I forget, something like 50 degrees.

MR. PETTI: Sixty degrees I think.

CHAIRMAN BLEY: Something like that, yes. But there's some odd stuff in there with some -- a large number I think of thermocouples failing and some other problems. Some tests that didn't come out quite the way you expected on the thermocouples. And it wasn't clear to me that your analysis included all of those kinds of problems. Did it? And can you point me to --

MR. PETTI: There's two ways to do the uncertainty analysis.

CHAIRMAN BLEY: Okay.

MR. PETTI: Let's say you could --

CHAIRMAN BLEY: How important is it, these results?

MR. PETTI: Right. So you could try to predict the thermocouple. The thermocouple is not in the fuel, it's near the fuel. And then say okay, if I hit that really well, let's say I miss it by 15 degrees, then I just look at the uncertainty from the thermocouple to the fuel. And I add the 15 degrees as a bias maybe.

We did not do that. We instead calculated it completely from the outside of the capsule in, first principles. So the thermocouples are there to help us control the experiment but the uncertainty analysis is a complete propagation from the outside in, all the effects that we know --

CHAIRMAN BLEY: Based on the physics.

MR. PETTI: Based on the physics. Now, AGR-1 happens to be unique because we borated the graphite to an extent that our gaps closed, tried to close. That made it very, very complicated. AGR-2, we don't see that behavior. I can show you graphite capsules that are just as complicated, some may argue more complicated. We can predict within 30 to 40 degrees.

So we're slowly building up. We've got, you know, I can show you a non-heat generating test graphite. I can take all the graphite irradiation effects, I can calculate that. Now with fuel I've got to add heat generation, I've got to add gamma heating of all the metals, all of that. That gets all folded in.

But given that there's still concern for the formal qualification we are going to do a mock-up, a heated mock-up. We also have three different ways to measure temperature now that we need to qualify in some way before we put them in the reactor.

We will basically put all of the different

TCs we're going to use in this -- we've got an actual sound, sonar-based wireless system that we're going to try. And we will set it up in the lab. We will change the gas mix like you do in irradiation and run the finite element model to show sort of as a way to qualify it before the irradiation.

CHAIRMAN BLEY: And that's in the continuing work.

MR. PETTI: That's in the continuing work, right.

MEMBER CORRADINI: And that's part of AGR -- the end radiation one is AGR --

MR. PETTI: The next one will be 5, 6, 7 altogether. So we're hoping to start that actually later this year.

CHAIRMAN BLEY: Okay. I'll try to study that. I'm a little vague on this stuff. I've been reading it, trying to --

MR. PETTI: AGR-1 has some oddities. I think when we do the results for AGR-2 it'll be better.

So in terms of the results in the high-temperature heating we have found that releases are very low and it's either fission products that diffuse in the matrix, a defective particle, and we have finally seen one or two particles fail at 17- and 1,800. We're not absolutely convinced yet till we do the full PIE but the odds are it looks like the particle fail in the reactor.

Now, just to tell you about how important this is though for 1,800 degrees. If we had taken German UO2 fuel and put it in the furnace at 1,800 degrees C we would start to see very rapid release after tens of hours. UCO TRISO, it just doesn't look like that. So while there are some failures the physics is definitely different. And we believe it has to do with oxycarbide fuel instead of UO2.

So that will all come out I think. We've got some UO2 in our second capsule. We plan to do some heating. But we're clearly beginning to see a difference between the two fuel --

MEMBER CORRADINI: So you intend to essentially do as you said, take the compact, go through the accident heating to show empirically what you would think is the root cause of what you get in terms of --

MR. PETTI: Right, right. Now we've got all this old German data that we can plot and show you the two different plots but we've got to go to the structural stuff.

MEMBER CORRADINI: -- staff would rather see.

MR. PETTI: Yes, we've got -- and it's not so much the heating as much as it is we've got to go in and do the detailed PIE which it requires some of these advanced techniques we have. I think I know what the problem is but again I won't go too far out on a limb. We think we know what the problem is with UO2. But 2 more years we'll know that answer.

So today you asked us based on last time the role of the reactor building in defense-in-depth and what's our approach to defense-in-depth. So we have presentations on those.

MR. ALBERSTEIN: Okay, my name's Dave Alberstein. I work for TechSource providing support to INL on the NGNP project.

And we decided to first talk about the role of the reactor building in defense-in-depth and reactor building design alternatives. We had a lot of questions in January regarding what if you do this to the design of the reactor building, what if you do that. What -- would a containment constitute defense-in-depth whereas other design alternatives you wouldn't have defense-in-depth.

So we decided that the best thing to do

before we talked about the general approach to defense-in-depth which Mark Holbrook will talk about would be to give you a rundown on the role of the reactor building in defense-in-depth and what happens in regard to offsite doses if you choose other design alternatives for the reactor building.

The slide that's up there right now, slide number 2, gives a quick review of the safety attributes, key safety attributes for the modular HTGR. These are listed because they have an effect on the decisions one makes with regard to how to design the reactor building. The fuel coolant and moderator are all chemically compatible under all conditions that we're aware of.

The fuel has large temperature margins in normal operation and during accident conditions. Normal operating temperatures are significantly below the temperatures at which significant degradation of coated particle integrity could occur. And the same is true during accident conditions.

Safety is not dependent on maintaining helium coolant pressure. If you lose coolant pressure you're not going to transfer large amounts of energy into the reactor building the way you might with other reactor designs that you're familiar with.

Post-accident heat removal, there's not a lot of heat that needs to be removed. In terms of the reactor building the RCCS removes the heat from the reactor system and does so passively. The response times of the reactor are long days as opposed to seconds or minutes as a result of the large amount of graphite in the core, its thermal capacity.

And lastly, there are multiple as we said concentric independent radionuclide barriers. Breaching the helium pressure boundary doesn't result in failure of the fuel or for that matter of the reactor building. So if we move to the next slide.

The role of the reactor building in safety design. The required safety function of the reactor building for the modular HTGR is to provide structural protection from both internal and external events and hazards for the passive heat removal of heat from the reactor vessel to the reactor cavity cooling system.

It needs to maintain the relative geometry between the vessel system and the helium pressure boundary and the RCCS. That's the safety-related role of the reactor building.

It does other things that are not required to meet the regulatory requirements for offsite dose. It does provide additional radionuclide retention.

And as was noted in one of the viewgraphs Fred presented you have to have that retention taken into account to meet the EPA PAGs at the exclusion area boundary. However, we do not need credit and don't anticipate that we ever will need to take credit for the reactor building's radionuclide retention capabilities to meet the regulatory requirements of 10 C.F.R. 50.34 for offsite dose at the exclusion area boundary.

The building also limits air available for ingress into the system after a helium pressure boundary depressurization. But this again is not needed to meet offsite dose requirements. Next slide.

MEMBER CORRADINI: You said a mouthful.

MR. ALBERSTEIN: Yes.

MEMBER CORRADINI: And so can we take the first bullet so I make sure I understand? So except for the fact that it keeps everything where it's supposed to be in the event of any sort of internal or external event you don't need the reactor building.

MR. ALBERSTEIN: To meet the regulatory requirements for offsite dose.

MEMBER CORRADINI: Okay, so I want to make sure I've got this right before we go on. So, I could have an open structure with just a bunch of steel girders to keep everything structurally in the same location and life would be fine.

MR. ALBERSTEIN: I have a slide on that in a little bit.

MEMBER CORRADINI: Okay.

MR. SILADY: The answer is yes.

MR. ALBERSTEIN: Yes.

MEMBER CORRADINI: Okay. The second bullet you said something, I think I should understand it but I'm a little bit confused. The EPA PAGs are essentially the limiting issue. So whether or not you meet 10 C.F.R. 50.34 is irrelevant because you want to meet the EPA PAGs to eliminate the need for evacuation.

MR. ALBERSTEIN: Yes.

MEMBER CORRADINI: But if you were not of so mind to do that and you were willing to come up with an emergency planning and evacuation scheme you wouldn't need the reactor building for that either.

MR. ALBERSTEIN: To meet the offsite dose requirements --

MEMBER CORRADINI: Of 50.34.

MR. ALBERSTEIN: -- 50.34, 52.79, that's correct.

MEMBER CORRADINI: I can't get all the

numbers in my head.

MR. ALBERSTEIN: Yes.

MEMBER CORRADINI: Okay.

MR. ALBERSTEIN: Yes.

MEMBER CORRADINI: All right. Then the only thing standing in the way of not having a reactor building versus steel structure is defense-in-depth which is you're putting all your eggs in the fuel vessel.

MR. ALBERSTEIN: We still need the reactor building to protect the structures that are needed for passive heat removal.

MEMBER CORRADINI: Okay, but that's not for fission product retention, that's for structural support.

MR. ALBERSTEIN: Correct.

MEMBER CORRADINI: So I'm still back to my original question which is you're -- and now maybe I'm misinterpreting. You're claiming I don't need defense-in-depth. The fuel is robust enough. As long as I keep everything where it's supposed to be through all the accident initiators in terms of structural dimensionality, that I expect this to stay here and that to stay there so that all those functions are met, I don't need a reactor building.

MR. ALBERSTEIN: To meet offsite dose requirements --

MEMBER CORRADINI: Right.

MR. ALBERSTEIN: -- I don't need a reactor building.

MEMBER CORRADINI: No. I wanted to make sure I got it right. Thank you.

MR. KINSEY: And again, to clarify, we'll talk further about the defense-in-depth aspects in a few minutes. We weren't covering that at the moment.

MR. ALBERSTEIN: Okay. Can we move on to the next slide? The reference design right now for the modular HTGR that was developed by General Atomics. And we're going to tend to talk in the context of that design. The next modular HTGR may be somewhat different. We can't speak for future designers.

But the reference design for that conceptual design was a vented reactor building. It addresses several specific design issues for modular HTGRs. Number one, it's compatible in terms of its volume with the fact that you have a non-condensing helium coolant that doesn't carry a lot of energy into the reactor building in the event of depressurization.

It's matched to the accident behavior of the modular HTGR. You vent it early in the transient when the radionuclides released from the helium pressure boundary are relatively low in terms of the activity. Later in these core heatup transients when the delayed releases take place the building is closed up where you can maintain some control over the rate at which those radionuclides could be later released.

And overall, providing this vented capability provides a more benign environment for the passive reactor cavity cooling system design, be it an air-cooled system or a water-cooled system. The vented building provides for lower heat pressure and structural loads on the RCCS all of which is advantageous in the design of the HTGR.

Next slide shows you a couple of things that are important. What this slide shows you are the vent paths that the helium would have to follow in the event of a helium pressure boundary breach to be released from the reactor building.

A couple of things to note. On the figure on the left at the top where it says "Operating floor elevation 6 inches," that's where grade is. So most of this reactor building is below grade.

Number two, note that the paths for depressurization and venting are somewhat tortuous.

It's not a simple matter of releasing directly to the atmosphere without going through a number of cavities that can have an effect on the amount of radionuclides that actually get out of the building.

And lastly, note at the upper left the final release point is about 20 feet above grade. So the points here are that the paths for release are tortuous. Most of the activity with regard to depressurization takes place below grade. The release point is at about 20 feet.

And then after the venting has taken place and the louvers are closed back up again most of the building is below grade. It's not sitting out there with the wind blowing on it and all of that. It's a below-grade structure.

So now --

MEMBER CORRADINI: I'm sorry. So this is from the `86 or -- I can't remember the date.

MR. ALBERSTEIN: That's late eighties.

MEMBER CORRADINI: Okay. That's the --

MR. ALBERSTEIN: GA design.

MEMBER CORRADINI: Okay. And then there was an upper bound calculation as to what would be the radionuclide release with that approach. And that was well within the EPA PAGs.

MR. ALBERSTEIN: I'm going to turn to Fred here since he did those --

MR. SILADY: Yes.

MEMBER CORRADINI: And staff issued a draft, a preliminary SER and concluded the same.

MR. SILADY: Generally. There were some asterisks --

MEMBER CORRADINI: What does generally mean?

MR. SILADY: There were some asterisks on the first NUREG-1338 that had to do with if the R&D was completed and so on. And I think the ACRS said it best in the wrap-up of that series of interactions is that neither the designers, the staff, nor themselves had found an event that exceeded the requirements.

MEMBER CORRADINI: And sorry. So to bring us back up to date now 25 years later or whatever it is, the concern about having dust reside in little nooks and crannies and that dust having radionuclides with it, does it change that conclusion? About dust transport during this. In other words, it's not just in the active -- it's not just in the pressurized helium but it's also in stuff that has eroded away from the graphite that's sitting in nooks and crannies that wouldn't have been cleaned up during the active cleanup during its circulation. That doesn't change the conclusion?

MR. ALBERSTEIN: The effects of dust need to be taken into account. And we believe that the extent to which dust effects can affect the offsite doses differs as a function of the design. In the case of the prismatics there's no historic evidence of dust buildup in prismatic HTGRs. In the case of pebble beds there is historical evidence of dust buildup and that would have to be taken into account.

MEMBER CORRADINI: Okay. The reason I'm asking the question is I've attended enough of the workshops that that answer tracks with what I've heard. But what I'm trying to get at is the uncertainty issue which is if I understand the system operation you're continually cleaning up the helium such that if God forbid you have an event like this there's not a lot of resident radionuclides.

Then the only place where you have a potential short circuit is these -- essentially an uptake of this and the dust is sitting somewhere within the system that you haven't cleaned up. It's not that I don't disagree with the logic, I want to understand how that affects the uncertainty in the estimate.

MR. SILADY: I think Dave made the distinction between the reactor types. And so we'll take it into account, whichever one.

It's not just the helium purification system captures the radionuclides or it's on dust. There's plateout on metallic surfaces. And it's not just that there's a leak in the helium pressure boundary and the helium leaves and the dust goes with the helium. There are a lot of other effects in terms of --

MEMBER CORRADINI: That I'm well aware of from other dealings. I'm well aware of that. On the other hand, all of those pieces of physics are tough to estimate. So I'm trying to understand if I had to draw an uncertainty bound on this, a range, that all the things that was concluded back in `86 still maintain even with the uncertainty you might get from it. That's why I'm asking the question. With prismatic. Let's not go back to pebble for the moment.

MR. PETTI: I think it's fair to say that.

MEMBER CORRADINI: Okay.

MR. ALBERSTEIN: There are, you may recall from the mechanistic source terms white paper there are design margins on both circulating activity and on condensable radionuclides like cesium, strontium, and so on and so forth. Between the best estimate of what circulating activities, for example, would be and the upper bounds that are assumed as initial conditions when an accident takes place.

The factor of 4 on the noble gases, the factor of 10 on the condensable radionuclides, the condensable metallics like cesium and strontium. And those design margins are intended in part to account for these types of phenomena.

MEMBER CORRADINI: Okay. So last question, then I'll stop. Is there anything from a testing or periodic maintenance approach to this that one could check to see that you don't have a buildup. Because I'm still -- I have this worry about this. And so if I can't be sure, and I have an uncertainty, is there something in periodic testing or maintenance that one can check this out?

MEMBER REMPE: Couldn't the staff suggest that a license condition be included, whether the instrumentation at the startup?

MEMBER CORRADINI: I guess I want to know -- I'm looking for some sort of --

MR. KINSEY: That's an attribute of course that would be specific to the design and it's something that I would expect that the design --

MEMBER CORRADINI: This is an instrumentation guru who wants to solve everything in instrumentation.

MR. KINSEY: That's right.

MEMBER CORRADINI: I'm looking for something that I can test or surveil on some periodic basis that gives me confidence that the uncertainty band is not here but it's there.

MR. ALBERSTEIN: Circulating activity and plateout activity can be monitored during operation. It was done at Fort St. Vrain. Will be, yes.

You're asking specifically about monitoring for dust levels and I'm hesitating just a little bit here.

MEMBER CORRADINI: I'm trying to come up -- well, I've made my point. I'll stop. I'm just trying to -- it just seems to me that that to me is -- this uncertainty is an important one relative to everything you're saying. Because if I buy what you're saying a lot of things naturally proceed from it. So I'm trying to find out what the uncertainty is on this.

DR. KRESS: I understand you have these measurements for the plateout and the circulating activity. My question is what are you going to do with them. Do you have some plans in mind, say, oh, well that's too much here, let's shut down the reactor and change out the fuel?

MR. ALBERSTEIN: At Fort St. Vrain, good example, there were actually tech spec limits on both circulating and some plateout activity levels. And if those limits were exceeded, yes, you'd have to shut the system down.

DR. KRESS: So you plan on doing the same sort of thing.

MR. ALBERSTEIN: I can't imagine not having tech spec limits on similar parameters.

DR. KRESS: I just never saw that anywhere in the white paper.

MR. ALBERSTEIN: Yes, and in fact we got an RAI early on from NRC on that subject. And we gave a lengthy response about what was done at Fort St. Vrain without making specific commitments for the guy that has to design and operate the next one. Because we just didn't feel given where we were with the design at that point in time and now that that would be appropriate for us to do. But historically the precedent is there.

DR. KRESS: It seems to me like in order

to put a tech spec on that you need to know a lot of information about how the events would -- during release plated out fission products on the primary system. I haven't seen much data on that. There's an -- how I say this but there's going to be some --

MR. KINSEY: It's in the plan.

DR. KRESS: It's in the plan?

MR. KINSEY: There are data. There will be more data. There's some data that you may not be aware of.

MR. ALBERSTEIN: Okay, let's move onto the next slide. We had a lot of questions in January as I said about what if you did this to the reactor building design or that.

What we're going to show you here to wrap this up are summary results of two reactor building design alternative studies that were done, one by GA back in the late eighties and one by the folks at PBMR roughly 10 years ago I believe, Fred? Yes, 10 years ago.

The first slide is a summary of some reactor building design alternatives that were considered by GA back in the late nineteen eighties. You can see five cases here that were considered.

The reference case, the vented building

with the moderate leakage rate of 100 percent per day. You can argue with the adjective "moderate" but that's what we used.

Other options that were looked at including putting filtration on the vents and still maintaining a 100 percent per day leak rate. Next, a filtered vent with a lower leak rate of 5 percent per day, that was option number three.

A couple of variations on option number four with a larger volume building to maintain relatively lower pressures during the depressurization event. And lower leakage, one variation was with an air-cooled RCCS, the other with a water-cooled reactor cavity cooling system.

Number five, two variations on another unvented, a larger volume so therefore lower-pressure design, one with a leakage rate of 5 percent per day and the other with a leakage rate of 1 percent per day. So these options looked at various combinations of filtration and building volume to assess what the effects on offsite dose would be of going with these containment design alternatives.

If you go to the next slide you see a summary of the results here. These are whole body doses at 30 days at the exclusion area boundary which was 450 --

MR. SILADY: Twenty-five.

MR. ALBERSTEIN: Four hundred twenty-five meters in this particular analysis against the frequency consequence curve that was in effect in the late eighties.

CHAIRMAN BLEY: I'm sorry, these dashed curves are --

MR. SILADY: -- cumulative distribution functions over all the events.

CHAIRMAN BLEY: And the 4A, 4B are different types of releases?

MR. SILADY: They're things on the previous page.

CHAIRMAN BLEY: Oh, the previous page.

MR. ALBERSTEIN: They're the alternatives.

CHAIRMAN BLEY: Okay, thank you.

MR. ALBERSTEIN: And you can see -- keep in mind number one, the number one there in the little square, the number one is the reference design well within the FC curve limits. And pretty much resulting in offsite doses that are equal to what was considered annual background back at the time this study was done which is a little bit higher today than it would have been back then.

You can see that the alternatives where one adds filters to the vents really didn't make much of a change in terms of offsite dose.

MR. SILADY: Not for the whole body.

MR. ALBERSTEIN: For the whole body. For the alternatives that have lower leak rates and higher volumes to result in lower pressure you did gain quite a bit on dose but you're gaining it relative to a point at which you weren't going much over background doses to begin with. So that raises the questions of adequate protection versus perfect protection for the public and whether an investment of resources to go to design alternatives such as numbers four and five are really buying anything of substance in terms of protection of public health and safety.

CHAIRMAN BLEY: Let me ask a question that's more aimed at Fred I think. I didn't ask this before because you weren't talking about it. Is there any requirement in the licensing framework for how far away from the requirements curve the PRA result CCDF has to form?

MR. SILADY: That's Tom's question earlier. And I'm not aware of it.

CHAIRMAN BLEY: Okay.

DR. KRESS: So if somebody wanted to add

two more modules --

CHAIRMAN BLEY: Okay, yes, it's a turnaround. Yes, it's a turnaround of the same question.

MEMBER CORRADINI: So I don't understand 4A and B. What does the air -- or the water RCCS do that 4A and B is different than 5A and B? Maybe I'm confused. Is it just the air and the water RCCS change the pressure?

CHAIRMAN BLEY: 4A/B and 5A are about the same; 5B is a little different.

MR. SILADY: They looked at it just to see if there was a discriminator there. Because when you go to a leak-tight building of any kind you can't have the air RCCS chimneys and so on, communication, the same way. So there's a different design on the air.

MEMBER CORRADINI: Oh. You're saying the plumbing is different.

MR. SILADY: Yes.

MR. ALBERSTEIN: If you look carefully here you'll see that 4A, 4B and 5A --

MEMBER CORRADINI: They're all the same.

MR. ALBERSTEIN: -- all the same. And in fact if you look at the previous slide they're all the same leakage rate of 5 percent per day. It turns out that's what's driving it. And then 5B, a little bit lower in dose because that was a lower leakage.

MEMBER CORRADINI: Oh, that was my interpretation. So the only difference between 1, 2 and 3. Well, wait a minute now. I wanted to finish that. So 3 is a different leakage but yet it is vented.

MR. SILADY: Filtered.

MEMBER CORRADINI: So that means what? I'm sorry that -- I'm looking at the same time to see if we have this 88.311 that I can look in detail. I'm sure we've got it somewhere.

MR. ALBERSTEIN: Filtration doesn't buy you very much for the whole body dose.

MEMBER CORRADINI: Okay. But -- okay. Yes, but the devil's in the details. What do you mean? You have the initial blowdown of the loss of pressure.

MR. SILADY: We don't filter it.

MEMBER CORRADINI: That's what I thought.

MR. SILADY: Yes. But you filter it after it closes.

MEMBER CORRADINI: So you've got some sort of dual valve. This thing blows down. The damper opens or whatever you call this thing, louver opens.

You let all that stuff out. This closes and everything seals up against that pressure and you filter everything beyond. Everything and beyond --

MR. SILADY: There's a filter in the reactor building and it's competing with the leakage from the reactor building.

CHAIRMAN BLEY: Just for me the case without the filter you talked about the tortuous path. That somehow you assume some DF on the tortuous path?

MR. ALBERSTEIN: Yes. It depends on the sequence.

CHAIRMAN BLEY: Or if you had a filter.

MR. SILADY: Yes, there's sequences with water, sequences that are dry and so you've adjusted the DF.

CHAIRMAN BLEY: Okay.

MR. SILADY: The point of the 4A, 4B and 5A were different designs of a leak-tight containment and we had to think about it broadly. What's it going to mean to our cost? What's it going to mean to our RCCS reliability? How is it going to work? And it ended up in looking at it only from the perspective of public safety it had the same curve. But looking at it from the perspective of cost or design margins and so on it had significant impact.

MEMBER CORRADINI: But just so I'm reading the numbers right, assuming all the numbers are right, the difference between 1 and 5B is an order of magnitude at low frequency and high dose.

MR. SILADY: About everywhere. It's almost parallel.

MEMBER CORRADINI: Yes. And the difference between 1 and 3 is about a factor of 2.

MR. SILADY: Yes, depending upon where you are because of the log scale.

MEMBER CORRADINI: And -- okay.

MEMBER REMPE: How much do you get from the deposition in the tortuous path?

MR. SILADY: It's nuclide-specific as you well know.

MEMBER REMPE: Is it -- raised it up to where you start getting close to your limits?

MR. SILADY: If there was no building you mean?

MEMBER REMPE: Yes.

MR. SILADY: If it was a reactor on the ground.

MEMBER REMPE: Yes. If you just released it out. Have you ever done a calc to see? I mean because then suddenly you are relying on the tortuous path for some sort of retention.

MR. SILADY: There's no way it's going to go over towards -- past the PAG. I mean we've looked at, you know, there's margin there.

We need additional tests and it's in the plan for radionuclide retention mechanisms in the building for our mix of helium and nuclides at these very small levels. Then we've got a whole lot of combinations of different releases. But surface deposition and so on is important.

MEMBER CORRADINI: So this is I guess, again, more details. And Dr. Bley will tell us to be quiet.

CHAIRMAN BLEY: We've already eaten about 15 minutes into lunch.

MEMBER CORRADINI: So is it already at the be quiet stage?

CHAIRMAN BLEY: It's pretty close, yes.

MEMBER CORRADINI: Okay, so one last question because staff brought this up. These are all blowdown and then long-term heatup where there is not a high point vent that I don't bring in air and continually heat up and oxidize. Is that correct?

MR. SILADY: These all -- all the ones that have a leak in the helium pressure boundary ultimately have some air that comes back into the reactor. Because it cools down. We're talking 30 days here.

MEMBER CORRADINI: I understand.

MR. SILADY: Okay. So they all have some of that. But there were no -- in this frequency range there were no leaks.

MEMBER CORRADINI: Where I had a low point enter and a high point exit --

MR. SILADY: No.

MEMBER CORRADINI: -- such that I could feed and continually oxidize.

MR. ALBERSTEIN: That's a 10 to the minus double digit scenario.

MEMBER CORRADINI: Okay. Okay, fine. I'll stop. Thank you.

MR. ALBERSTEIN: Let's move onto the next slide. As I said, also there were studies done in the last 10 years or so by the folks at PBMR and Westinghouse on alternative reactor building configurations for the pebble bed.

You can see here the list of alternatives that were examined. Again, unfiltered, vented with moderate leakage as the reference case. Case 1B, adding blowout panels between components within the reactor building.

The next case, option number 2, partial filtering. Venting again with blowout panels and a moderate leak rate.

Then option 3A, a lower leak rate in the range of 25 to 50 percent a day, again filtered, fully filtered in this case with blowout panels again.

3B, adding an expansion volume for the gas to the building. And then options 4A and B, looking at a pressure-retaining system with internal blowout panels and very low leakages, less than 1 percent per day.

So again, a broad spectrum of alternative designs were considered and the next slide shows you the results. I said we had a slide on no reactor building.

What we have here is a comparison of thyroid dose at the exclusion area boundary against the EPA PAG limit on thyroid of 5 rem. And you can see in the case of alternatives 1A and 1B which is the second column from the left there was substantial margin relative to the PAG. So with alternatives 2 or 3 did further reduce the dose and increase the margins relative to the PAGs.

CHAIRMAN BLEY: And according to our

previous discussion these are mean values.

MR. SILADY: Yes. Yes.

MR. ALBERSTEIN: Yes. However, in some cases added features can fail. And if that happens late in a sequence when the delayed release is taking place the gains in margin relative to alternatives 1A and 1B are lost. And that's what you see in the crosshatch columns.

The pressure retaining design, 4A, also increased margin relative to the PAGs. But again if late in the sequence you have late failure due, for example, for a seismic aftershock you can actually get higher doses offsite for the thyroid than you would get from the base case alternative. And in fact higher doses than you would have received if you'd had no reactor building at all.

So the bottom line here is that one has to be careful when throwing around ideas with regard to reactor building design for modular HTGRs. You have to be careful to understand the accident behavior of the HTGR system. Bottling things up isn't necessarily going to buy you the reductions in offsite dose consequences that you would intuitively think you would get.

MEMBER CORRADINI: But to get back to

Joy's question about the previous study, am I reading this right, that I have a DF of 20 if the reactor building is there?

MR. ALBERSTEIN: Which column are you looking at?

MEMBER CORRADINI: I was looking at the purple that's 10,000 and the next purple over which is 500. That's a DF of 20.

MR. ALBERSTEIN: For iodine.

MR. SILADY: For this event with thyroid.

MEMBER CORRADINI: Which is the limiting -- based on something I thought you said earlier that's the limiting thing for the EPA PAGs, right?

MR. ALBERSTEIN: We think that's the harder one to meet from the whole body, yes.

MEMBER REMPE: So from what you're saying, earlier you said well, it's only safety-related for heat transfer. But then there's certain criteria associated with this building for decontamination and reduction and release that is relied upon. So it seems like --

MR. ALBERSTEIN: To meet the PAG.

MEMBER REMPE: Right.

MR. ALBERSTEIN: To meet the PAG but not to meet the 25 rem whole body requirement.

MEMBER REMPE: But since there's no evacuation planning associated with this approach it seems like there would need to be some sort of regulatory oversight in the design of the building to make sure that you don't need evacuation.

MR. SILADY: There obviously would be a regulatory oversight on the entire plant.

MEMBER REMPE: But there's to be some criteria like tech specs or something.

MEMBER CORRADINI: But can I try her question differently? The building -- I mean I'll go back to the previous one where you said it's a factor of 2. So, let's say they're all the same within the uncertainty of all this sort of stuff. That means the building is a factor of 20, adding a filter is a factor of 2 so I've gone from a factor of 20 to a factor of 40 to 50 decontamination factor for the very fact of the presence of the building, it in itself is the filter. So you would have to have some sort of performance objective for the building to perform as a filter, otherwise you get -- you go beyond your limit.

MR. SILADY: The reason that you see the difference between the two first two columns is no building, not even a tent, doesn't quite meet the PAG.

We're getting tremendous radionuclide retention within the fuel to be even that close.

MEMBER CORRADINI: I'm not following.

MR. SILADY: So now you go to the second column and we say whoa, based on our best estimates of the DFs this is sizable. We need a little more data here to see if that 20 is real.

MR. ALBERSTEIN: And there's no denying that a building versus a tent helps.

MEMBER CORRADINI: I know, I know. But I'm just, I'm framing it this way because there were some bullets somewhere in one of these presentations that it's strictly structural. My point is this at least demonstrates by sensitivity it's not strictly structural. It in and of itself has to have a performance objective because it's performing as you call it a leak in containment, we'll call it a confinement, whatever you want to call it, that you now have to show performance on. Otherwise it doesn't meet your objective.

MR. SILADY: The objective we're talking about here is the design target.

MEMBER CORRADINI: Yes, I understand.

MR. SILADY: Okay.

MEMBER CORRADINI: I'm with you.

MR. SILADY: So the structural -- and the reason to make it safety-related was for 10 C.F.R. 50.34. That was where that statement was.

It's clear that -- we had it as a bullet. To meet the PAGs we need the building.

MEMBER CORRADINI: Okay. All right.

MEMBER REMPE: But it can't just be any building is what I'm --

MR. SILADY: Right.

MEMBER REMPE: It's got to be a certain.

MR. SILADY: Yes. And PBMR had a different looking building than the MHTGRs, much different. Largely above grade.

MR. ALBERSTEIN: And their building, a decontamination factor of 20 is a little bit higher than what the MHTGR folks assumed.

CHAIRMAN BLEY: I think we can move on and close this out as quickly as you can. Give Mark some time although he's got enough slides to go for an hour it looks like.

MR. ALBERSTEIN: I won't. In summary with regard to reactor building alternatives. We believe that the vented building, the reference design is the best match for the characteristics of the HTGR.

For the low-frequency events a high-pressure low-leakage LWR-type containment can actually increase radionuclide release relative to our reference case. You want to add filters or active HVAC systems you can get some small improvement in offsite dose. But under certain low-frequency event scenarios they may not be available and again provide you relatively little additional margin.

So these studies really have confirmed the decision in our design approach to place the emphasis on retention at the source within the fuel. There's a whole lot more detail on all of this in response to the RAI number FQ/MST-82 and all the references that were provided with it. That is all I have.

CHAIRMAN BLEY: Thank you. Mark, how long do you think it'll take to get through yours? The ones that -- there's some that are almost repetitive.

MR. HOLBROOK: Yes, exactly.

CHAIRMAN BLEY: So focus on the things that are new that you're trying to tell us from the last meeting.

MR. HOLBROOK: My name is Mark Holbrook. I work for the Idaho National Laboratory. Been involved with the NGNP project for some period of time.

And as was mentioned by the chairman we

have several technical points in these series of slides that have been brought up before by both David Alberstein and by Fred Silady. And so we'll move through some of those technical points and try to focus more on the higher-level structure of how we want to evaluate defense-in-depth as an approach.

Slide number 2 provides an overview of where we're headed in this presentation. So we'll kind of move on. We're going to discuss very briefly on the next slides a few details and then we'll get into the bulk of the presentation which is in the center of this slide, plant capability, programmatic and risk-informed evaluation of defense-in-depth. Then we have some summary slides at the end.

The overall intent of the approach is to develop a structured system for evaluating defense-in-depth adequacy. So we want to be able to define how we're going to evaluate whether we have adequate levels of defense-in-depth in the overall design and the process that would be applied by a future applicant.

The elements involved in this evaluation process would be looking at the plant capability to provide defense-in-depth, to be able to look at the programmatic elements of defense-in-depth that would be applied by a future applicant and to look at the risk-informed evaluation process to be able to determine whether we've met all the principles that we've laid out in our white paper for defining defense-in-depth.

If you look in Chapter 2 of the white paper there's several discussion points and references from the regulations looking over a long period of time of how people have tried to define defense-in-depth in the past. That paper was written approximately 2 years ago so we thought it would be more useful maybe to look at something more recent such as this summary of NRC's defense-in-depth strategies you see at the bottom of the slide.

This comes out from last year's order for the containment fence that was issued on March 12 of 2012. And you can see it on the screen there but it focuses on the definition of defense-in-depth in the context of prevention, mitigation and emergency planning. So at the end of these series of slides I'm presenting today I'm going to recast what our process looks like within that structure.

The next series of slides talks about those three principal elements, plant capability, programmatic and evaluation. On this first slide here we focus first on plant capability. In fact, the next series of slides draws in many of the physical points that have been made in the other presentations having to do with the attributes of the plant that provide defense-in-depth.

So you see there there's kind of a triangle chart or a graphic that you might look at that tries to pull together the concept. You see down in the lower lefthand corner plant capability, defense-in-depth, in the lower right programmatic and then risk-informed evaluation with a triangle in the center that talks about the PRA results and the deterministic analysis.

We're first focusing on that lower left triangle, plant capability, defense-in-depth, which reflects the decisions that are made by the designer to implement function structures and the SSC design and availabilities to ensure that we have defense-in-depth in the plant.

So a lot of the things that have been talked about in previous parts of today's presentation, the inherent reactor characteristics that we take advantage of. Event progression, time cost is provided by the graphite in the core. We're going to focus a little bit in the subsequent slides on the radionuclide barriers to prevent release of radioactive material. How the passive and active SSCs work together to provide us defense-in-depth. We also implement some of the discussion that we've had in previous meetings about SSC safety classification, design margins, conservative approaches. All these things are factored into what we would call plant capability defense-in-depth.

If you look at the reactor nuclide barriers we've mentioned in previous points in today's presentation that the barriers are concentric, they're independent, and that their performance emphasis is on the performance of the fuel barriers as we look at those.

Also as mentioned previously the reactor building provides defense-in-depth for meeting the top-level regulatory criteria at the EAB. However, as was previously noted we do need to rely on the building to meet the PAGs at the EAB.

Both active and passive SSCs are working in concert with these inherent design characteristics to reduce the frequency of challenges to the radionuclide barriers. And we'll talk about a series of challenges in the subsequent slides that Fred brought up in his presentation first thing this morning.

But our process is looking at a full spectrum of events, not just a few limiting or bounding events. So we're looking at all challenges to barrier integrity and independence. So we're looking at all the possible failures that could be within the frequency range domain that we've defined in our licensing basis event process and we're taking into consideration all of those things.

Safety margins and conservative design approaches will be used to address uncertainties in barrier and SSC performance. And this refers back a little bit to some of Tom's earlier questions about how we would consider the reliability and the availability of the SSCs as part of our event selection and our overall approach to defense-in-depth.

Fred mentioned this morning three key functions that we've got to maintain in the plant to be able to minimize the challenges that we make to our integrated set of barriers.

The first one is control of heat generation. We have a combination of inherent reactor characteristics which is a plant capability issue. And the available SSCs both passive and active available for our consideration when we look at control of heat generation.

So in this case as Fred mentioned earlier we have two independent and diverse systems for reactivity control. And they rely on gravity. So in a loss of power control rods will shut down the plant. We also have an independent system, the reserve shutdown system that Fred mentioned earlier. Each of these systems is available for maintaining reactor subcritical. Each of these systems available for cold shutdown during refueling. So that's one aspect of plant capability.

Another key function that we have to maintain is removal of core heat. In this case here we have a combination of both active and passive systems. Again these have been mentioned prior.

The normal mode of removing heat from the plant of course is to use a helium transport system to be able to remove that heat out of the plant under normal operation to the steam generator and finally to the ultimate heat sink.

But we also have alternate methods. We have a shutdown cooling system that is an active system. It's typically not -- we don't see it as being safety-related but that is available during shutdown periods and during planned maintenance to be able to remove the heat from the helium system. Again, this system that draws the helium through a heat exchanger which then has its own cooling water to be able to provide the ultimate heat sink.

However, we also have a passive system that's available for off-normal events, provides heat removal and investment protection. This is the reactor cavity cooling system, RCCS system. In that mode of operation we have the heat being radiated from the reactor vessel which is uninsulated to the panels that are surrounding the reactor. And then again we have either a passive air system or a water system available to remove the heat from the RCCS.

Any questions on that?

The third mode, third function, control of chemical attack that Fred mentioned this morning. In this case we have a combination of inherent reactor characteristics and design features that minimize the effects of chemical attack.

As far as inherent characteristics you can see on the list there we have non-reacting helium as a coolant. We have slow oxidation rates afforded by the graphite that we have in the core. And again, it was mentioned before, the water-graphite reaction is endothermic and we need to have very specific conditions to be able to even postulate the kind of sustainable reaction rate within the core.

From a design feature we have limited flow area through the core. It was mentioned we have a large L over D on this plant. There is a lot of flow resistance in the coolant channels.

Reactor building is embedded so that tends to minimize some of the effects of leakage from the building. We also have limited sources of water. We have moisture monitor, steam generator isolation and dump systems to try to minimize the injection of water into the plant. Again, these are all design features.

And then again when you look at the fuel itself within the particles, within the compact, within the matrix we have several layers that if we did have an oxidation event it takes a long period of time before you could postulate that you would be able to have a direct attack on the particles themselves from an oxidizing event. So these are all meant to address some of the plant capability aspects of defense-in-depth.

The second element, programmatic defense-in-depth, is all those processes and procedures that you would expect to be in place at any operating plant. Inspection programs, test programs, quality assurance programs, all those kind of things would be implemented at this facility. You see some examples there.

Special treatment requirements in particular. Some of the questions we had earlier about technical specifications. Yes, we understand there would be some design limit specifications on the fuel. Also, it's reasonable to expect that we'd have some circulating activity requirements that we need to implement through tech specs. All these kind of things would be implemented. And that would fall under programmatic defense-in-depth.

Now, the final part of the triangle is the risk-informed evaluation of defense-in-depth. This third element provided by the evaluation process that you see that's fed by information that's coming out of the PRA and also from deterministic safety evaluations that you would typically do in Chapter 15 for your plant provides a framework for performing these evaluations to determine how well your plant capability and programmatic strategies are being implemented.

So it provides accident prevention and mitigation insights, it provides input into your safety classification to -- when you have several sets of SSCs that are available to provide required safety function it allows you to choose wisely in the selection of those particular systems to provide the safety functions that are needed. And it also provides an opportunity to identify key sources of uncertainties.

So on the next slide we talked a little bit more about risk-informed evaluation. We want to identify credible failure modes and challenges to the barriers including dependencies and interactions along the barriers and other SSC failure modes. This is what I mentioned earlier is we want to challenge our design and make sure that our PRA is looking at all the possible spectrum of events and failure modes that need to considered.

We also identified the roles of SSCs in this process in the prevention and mitigation. We wanted to make sure that we have a balance of prevention and mitigation in our design. We want to quantify the extent to which accidents are being prevented and mitigated. And we're using both the PRA and our safety evaluations to do that.

And finally, we want to establish that there are no events with significant frequency of occurrence that rely on a single element of the design or programmatic approach in protecting the public from a release.

So again, the whole purpose of the risk-informed evaluation is to be able to scrutinize the effectiveness of both the plant capability of defense-in-depth and the programmatic defense-in-depth.

In fact, this approach was called out in Appendix C of NUREG-2150 which is the Risk Management Task Force where they mentioned the fact that this particular process that the NGNP is proposing includes a concept for using risk assessment methods as a measurement of effectiveness. And they called that out as an paragraph that is very similar to the approach or the thinking of the task group.

MEMBER STETKAR: Mark, before you leave this, and I hate to do this because of time so I'll try to make it quick. You've heard a couple of questions already this morning about concerns in terms of evaluating the margin to the acceptance criteria.

And I'll just read something out of the defense-in-depth paper. It's in Section 3.3.2.2 if you want to look it up. It says, "If the 95th percentile of the frequency of the licensing basis event is above the break point for separating the AOOs from the DBEs, or the DBEs from the BDBEs the licensing basis event is assigned to the higher frequency category where more stringent dose criteria apply."

MR. HOLBROOK: Yes.

MEMBER STETKAR: Vertical scale. The 95th percentile from the consequence uncertainty distribution is required to be within the associated frequency consequence curve. That is a statement. That is a measurable metric from my risk assessment to give me confidence in the margins regardless of whether I'm doing a single unit or a multi-unit plant in consequences.

That's from a December 2009 paper. By the time we get to the licensing basis event selection and everything else we heard we've abandoned that notion of 95th percentile over the whole frequency consequence curve. So I'm curious about why it's morphed into that.

And I'll ask the staff more about that because you pointed that way. Because when I read the defense-in-depth paper I said geez, I understand how they're doing this. I understand now how I can quantify my confidence in those margins both vertically and horizontally, wherever I am on that surface. And now today I can't understand how I can quantify that confidence in the margins.

MR. HOLBROOK: It's certainly true within the DBE region. Okay. Your question really --

MEMBER STETKAR: It's over a whole surface. I'm interested in the whole surface.

MR. HOLBROOK: -- the AE region and the beyond design basis event region.

MEMBER STETKAR: And with that because of the time I'll let you finish.

CHAIRMAN BLEY: Kind of -- if I can rephrase. If that one makes sense to you then why aren't they all the same?

MEMBER STETKAR: Exactly. Yes. Because if they're all the same then I understand how I can measure my confidence in my margins. I might disagree in terms of how confident I should be, whether that should be 99th percentile or 90th percentile. But at least I can measure it.

DR. KRESS: That depends on how good you know these probabilities.

MEMBER STETKAR: That's okay.

CHAIRMAN BLEY: But all the rationales for all that discussion is in the licensing basis event. And Fred regurgitated that this morning.

MEMBER STETKAR: I understood the rationale -- I understood all of the rationales on the vertical axis. I didn't understand them on the horizontal.

MR. HOLBROOK: This slide here is just to provide an integrated picture of some of the details that I presented on the previous slides. So it's kind of a takeaway slide if you want to look at this construct and refresh your memory on what pertains to what, and what insights are flowing in what direction and all of that.

Is there any questions on that?

CHAIRMAN BLEY: I'm trying to remember. At the last meeting your approach framework for defense-in-depth is very broad. Kind of, almost everything in the design is focused through defense-in-depth. I think some people objected to that and were asking you hard questions about it.

MR. HOLBROOK: I would characterize it a little bit different. What you see in plant capability defense-in-depth and what you see in programmatic defense-in-depth are the kinds of things that you would expect to see when you try to define defense-in-depth for today's light water reactors.

I think the thing in my opinion that is

somewhat more new is to then use the probabilistic risk assessment process and the safety evaluation process for your accidents in Chapter 15 to go back and reassure yourself since we have the opportunity to do this during the design phase, to go back and convince yourself by going through additional steps that we haven't presented in these slides but are found in the defense-in-depth paper such as Figure 3-7.

To systematically go back and look through all the principles that we defined in that white paper to convince ourselves that the plant has adequate -- the physical plant has adequate capabilities and we also have programmatic where needed and that there is a reasonable balance between the two.

Not to have an over-reliance on programmatic to cover up for some deficiency in the plant. We want to have a balance and we want to have a structured approach which is really what this is doing is a structured approach to go back and systematically convince ourselves that we have those elements that are traditionally found in light water reactors.

That's really what this process is all about and I think that's what they were trying to say in NUREG-2150, that we have -- the thing that's different here is (a) we're doing it up front, (b) we have a structured approach that uses risk and determinism to be able to evaluate whether we have adequate defense-in-depth before the plant is ever built. Does that help?

MEMBER STETKAR: That's good, thanks.

DR. KRESS: The light water reactors generally have diesel generators and batteries to guard against loss of offsite power. You don't have any of that here, right?

MR. HOLBROOK: Well, it depends -- well no, the answer is no because again it comes back to your plant design specifics about what is your reliance on 1A with AC distribution. If you have passive safety systems that don't rely on that then you don't need it. Okay? So it's design-specific. Okay?

In summary, what I wanted to do here, we've got this slide then we've got one final slide that kind of sums up everything that's gone on today.

What we wanted to do here is we wanted to just translate what I talked about, some of those characteristics within the context of prevention, mitigation and emergency planning such as the definition that you found on slide 3 I believe it was.

So again we're trying to evaluate the plant capability and programmatic elements with an integrated risk management approach. I just explained that.

Within the context of prevention, mitigation and emergency preparedness again there is several aspects and I didn't try to list them all here because I wanted to keep this down to one slide, but there's plant capability aspects that fall under prevention, there is certainly plant capability aspects that fall under mitigation, and then there's also administrative or programmatic I should say elements that fall under mitigation and also emergency preparedness.

All we tried to show is that we're looking at these things in a similar manner as the staff would look at things, looking for prevention, mitigation and emergency preparedness issues. But we're trying to do it in a structured, integrated, risk management approach. So, that's all I was trying to show. This slide, there's really no new technical information on that slide besides what we've already discussed.

Then finally, this is the last slide in our presentation this morning. Again we wanted to leave you with a takeaway looking at -- maybe at the 10,000-foot level of some key attributes that we've discussed at several points this morning. Our design, our approach addresses the full spectrum of internal and external events on a per-plant year, not on a per-reactor basis.

We've mentioned several times it includes events that could affect multiple reactor modules to be able to assess an integrated plant risk. So we're looking across all of the plants and that -- or all the reactor modules that would constitute a plant to ensure that any events that would affect multiple units are taken into consideration and are included and would be displayed on a frequency consequence curve.

Again we mentioned we're using ceramic fuel. It doesn't melt when challenged by a full spectrum of internal and external events.

We are looking at cliff edge effects. We want to find that safety terrain, make sure it's adequately addressed. So we're looking at events below 5 times 10^{-5} down into the 10^{-8} range to be -- make sure that we don't miss something that we need to assess in our process.

We assure that the safety is not wholly

dependent on any single element. Again, this is part of our defense-in-depth that I've just gone through. We want to make sure that we don't have any single point kind of failure whether it's in the design, construction, maintenance, or operation of the facility.

We want to provide compensatory means to make sure that prevent accidents are less than the effects. In other words, a balance between prevention and mitigation if a malfunction occurs.

And finally, as we mentioned several times we have multiple, concentric, independent radioactive nuclide barriers that we want to protect. A breach into the helium pressure boundary does not result in failure of the fuel or the reactor building. In other words, we don't rely on the presence of helium to be inside the core to be able to get heat out of the core as we mentioned during some of the earlier slides in the presentation.

So with that our presentations are finished. We thank you very much for your consideration. We'd take any other questions that you may have.

CHAIRMAN BLEY: That's great. Thank you and thanks for doing that quickly and effectively.

We'd like to mention on the record that Dick Skillman, member of the ACRS, has joined us during the morning.

We're going to take a break now for lunch. I'm sorry, we are going to come back at 1 or try to as close to that as we can because we have a full afternoon. We could have started earlier but we thought the agenda this morning would not take as long as it did. And we'll recess until 1 o'clock. Thank you all.

(Whereupon, the foregoing matter went off the record at 12:16 p.m. and went back on the record at 1:00 p.m.)

CHAIRMAN BLEY: The meeting is back in session. We expect we'll have some more members this afternoon as other meetings finish up but we have no one yet. In fact we're missing two. I think Joy is gone. Joy's gone, that's right. But Mike said he'll be back after the lunchtime meeting. Sam is also gone, same meeting. Sorry for the mumbling. I should have done that before I opened.

At this time we're looking forward to hearing from staff about their evaluation of these white papers we've been hearing about at our last two meetings. I will turn the meeting over to Anna Bradford at this time and we look forward to hearing from the staff.

MS. BRADFORD: Thank you. We appreciate being here today. My name's Anna Bradford. I'm the chief of the Small Modular Reactor Licensing Branch II in the Office of New Reactors. And we're here today to talk to you about some work we've been doing over the last few years for the Next Generation Nuclear Plant.

As you heard from DOE we've been focused on some very important issues such as mechanistic source term and event selection. And some of these issues are being addressed in a broader sense in other activities in the Agency such as Fukushima-related activities or the small modular light water reactor licensing activities. But I just want to point out that today we're here to specifically talk about the NGNP issues and that design-specific information and the current regulations as they apply to what we've been thinking about.

So we're meeting with you today. We're on the schedule to meet with the full committee in May. After that we're hoping to get a letter from the committee with comments that you may have on our assessments and then we'll finalize our assessments. And those will be sent to DOE and also made publicly available.

So we look forward to the interactions today and at this time I'll turn it over to our senior project manager Dr. Don Carlson.

CHAIRMAN BLEY: Thank you. Don?

DR. CARLSON: Good afternoon and thank you committee members for this chance to present our assessment findings, our results for the NGNP pre-application review activities on these key licensing issues.

Again, my name is Don Carlson. I'm the PM for NGNP and before the break this afternoon I will be joined by Tom Boyle and Jonathan DeGange. And after the break I'll be continue with help from Jim Shea and Arlon Costa.

We have -- my presentation is really just an overview, an introduction. So big questions about our findings really should come in the presentations that follow my overview. So then I will turn it over to in the second part of the agenda to my partners here at the table to talk licensing basis events and then Jim Shea on source terms. I will give the talk on functional containment performance and Arlon Costa on emergency preparedness. So you have our contact information there.

We've had many contributors to this activity over the years. Sud Basu has been involved in these activities from the very beginning as have I. Mark Caruso, Michelle Hart. We have in the appendices of two of our documents a list of contributors to these activities. Stu Rubin who contributed early on to these activities and has come back as a member of the public after 1 and a half years of retirement, he's in the audience today.

CHAIRMAN BLEY: We'll look forward to hearing his comments.

DR. CARLSON: I also have on the phone today Mike Kania. Dr. Mike Kania is an expert consultant on TRISO fuel. He has been a major player in TRISO fuel R&D during his Oak Ridge career in the seventies, eighties and nineties and has collaborated extensively with the German TRISO fuel program over the decades. Now he is working through Brookhaven National Lab and has been a major contributor to our recent activities on fuel qualification for TRISO fuel. So I've asked him to listen in on the phone and stand by to help us discuss detailed questions that any members may have.

So as Anna mentioned we are requesting a letter and are on the full committee schedule in May to do that.

I'm going to give a little overview now of the project history. You're probably familiar with a lot of this but the NGNP project was established by the Energy Policy Act of 2005. DOE and INL were tasked to demonstrate by 2021 a prototype high-temperature gas-cooled reactor for co-generating electricity and process heat. The NRC has licensing authority for the prototype plant.

So as stipulated by the EPAct the DOE and NRC jointly issued a licensing strategy report to Congress in 2008 and selected -- jointly selected an option 2 where option one would be a traditional deterministic approach and option 4 would be a more -- quite a risk-based approach. And so option 2 and option 3 would be those two extremes.

And so option 2 is described as a risk-informed and performance-based approach where you use deterministic engineering judgment and analysis complemented by PRA insights to establish the NGNP licensing basis and requirements.

So the other major activities we've been going through are what we are talking about today, a series of white paper submittals that we have been assessing since 2010. A year and a half ago DOE with

-- based on a review by the Nuclear Energy Advisory Committee, some of whose members are here on the ACRS, decided in a letter to Congress that DOE will not proceed with the detailed design activities at this time and instead will continue to focus on high-temperature reactor research and development.

And the key item here, interactions with NRC to develop a licensing framework. And then establish a public-private partnership, and that was the part that has really been difficult for them to establish and go forward with the plan.

So the plan was, as you may recall, to meet the 2021 demonstration target to submit an application to us by 2013, this year. And we have not been on track to do that.

So we have been using DOE-reimbursable funds here in the NRC to engage within DOE on these four key -- using four key areas. And I'll discuss what those are in a minute.

So we issued our preliminary assessment reports to DOE in February of last year. And as was indicated today there was an extensive RAI process, request for additional information, hundreds of those. And I would like to acknowledge that DOE and INL did a very thorough and prompt job of responding to those

RAIs.

I'd also like to say that the staff provided some very insightful questions and comments to go with those. Together all those RAIs and RAI responses are a substantial repository of what we've learned over the last few years.

And if you look at the page count they actually far outnumber the pages of the white paper we're reviewing. So I would urge future people engaging in follow-up activity of this kind to study all of those materials, not just what we're presenting today.

So the products that we issued last year were basically Rev zero of the initial assessments of the fuel qualification white paper and mechanistic source terms white paper, and then another report assessing the contents of what we called the set of white papers describing their risk-informed performance-based approach. And those were the defense-in-depth white paper, the LBE licensing basis event selection white paper, and the safety classification of SSCs white paper.

Then also in February NRC issued a letter to DOE agreeing to focus on four issues in these four key areas, licensing basis event selection, source terms, functional containment performance and emergency preparedness.

We started engaging and then in July of last year DOE provided a letter to us that clarifies their approaches to the issues and exactly what staff feedback they would like us to provide on those issues.

And so we engaged in a number of public meetings and conference calls through November of last year. And we also reviewed a number of supporting technical documents that DOE and INL submitted during that time that clarified some of these issues. Again, as you know in January DOE provided an information briefing.

And now let's talk about the three products. So again what we are asking the committee to look at and write a letter on ultimately is these three staff products.

Product one is what we call the issue summary report. Its formal title is "Summary Feedback on Four Key Licensing Issues." And those are the four key issues that we talked about just now.

And then Rev 1 of the two assessment reports. So an updated assessment report on fuel qualification and mechanistic source terms and an updated assessment report on the risk-informed performance-based white papers with additional insights from the PRA white paper that we reviewed during this time.

This update, the Rev 1, the things that we've been doing since February involved additional staff beyond the staff that were involved in the initial phase activities that gave us the Rev zero of these reports. And so we have different perspectives and high-level concurrence on the staff positions presented in these updated papers.

So let's talk a little bit about this issue summary report. Again, the four issues. These same four issues were highlighted in the joint DOE-NRC licensing strategy report to Congress in 2008.

The same kinds of issues were considered in NRC pre-application activities for modular HTGRs. They have been packaged in various ways but they always come down to these four major issues.

And as we noted there was a lot of engagement on these issues in the late eighties and early nineties with DOE and General Atomics for the MHTGR and in about 10 years ago on the pebble bed modular reactor. In fact, the approaches proposed by DOE and General Atomics back then for modular HTGR and for pebble bed modular reactor are very similar to what we now see for NGNP in that we have similar FC curves, the event frequencies are always per reactor-year so inherently account for multi-module effects.

And it was interesting to note that what we now are wanting to call beyond design basis events were at that time called emergency planning basis events. So the terminology has changed a little but there's a lot of similarity in the approaches.

So all of the issues that we are talking about we have developed our feedback in view of all relevant prior staff positions and all ACRS comments on those staff positions and Commission direction on these issues in various SECY documents. Starting most notably with SECY-93-092 and the NUREG-1338 which documented the preliminary safety evaluation of the MHTGR.

And then more recently SECY-03-0047 which was about the pebble bed reactor review, pebble bed modular reactor, SECY-05-006 which was an information policy SECY paper that talked about modular HTGRs but also technology-neutral framework. And then NUREG-1860 which was about the technology-neutral framework. And a year and a half ago SECY-11-0152 which Arlon Costa will be talking about later in the presentation on emergency preparedness.

DR. KRESS: Are all of these available from ADAMS?

DR. CARLSON: Absolutely. And I think you have links to all of them that I provided to Maitri.

Another point to emphasize is that these are risk-informed performance-based approaches that have been proposed for NGNP are similar to approaches that have been or may be considered for NUREG-1860. NUREG-2150, a Near Term Task Force recommendation, et cetera. So a revised -- a new framework resulting from all of these other efforts may very well change the staff positions that we're describing today for NGNP.

Again as Anna emphasized we're not talking here about any of those other efforts. We're talking strictly about our evaluation of the proposals for licensing a modular HTGR, namely the NGNP.

Now, Dr. Corradini had a question earlier about how this relates to the notion that we were going to do a pilot study test-driving the technology-neutral framework NUREG-1860. And this is --

MEMBER CORRADINI: Am I remembering some Commission directive incorrectly?

DR. CARLSON: You're remembering it very

correctly I believe.

MEMBER CORRADINI: Okay, all right.

DR. CARLSON: And what we said in that was -- what we were told to do and what we committed to do was to do a pilot study of that type of approach in parallel with adapting existing regulations as described here.

And so did we start doing that? No. Would we start doing that sometime before expecting to receive an application like this? I believe we would. So.

CHAIRMAN BLEY: That's still in the plan for expanding the guidance.

DR. CARLSON: That's what we committed to do and until we have a different commitment that remains what we --

CHAIRMAN BLEY: But it has not begun.

DR. CARLSON: It has not begun because there's nobody -- no scheduled for getting an application for NGNP or any other gas-cooled reactor.

CHAIRMAN BLEY: Can't give the example without a design.

MEMBER CORRADINI: So what would -- you described it well. So what would trigger you starting planning to do that? Somebody actually putting in a pre-application intention with you guys?

DR. CARLSON: The notion of course for a technology-neutral framework is to be able to handle any technology so it would be applicable to a liquid metal-cooled reactor.

MEMBER CORRADINI: Oh no, I understand that.

DR. CARLSON: So right now --

MEMBER CORRADINI: Planning to do it would occur when?

DR. CARLSON: I would say a few years before we expect, you know, 2 to 3 years before we expect to get an application, something like that.

CHAIRMAN BLEY: Certainly after you have a design.

MS. BRADFORD: If I could just clarify for one second. I think what you're talking about is a SECY paper from a couple of years ago where we said we would do a pilot study of a technology-neutral framework for the HTGR, specifically the NGNP design. And in that paper we said following submittal of the NGNP design application the staff would conduct a limited comparison study of the application in parallel with our review of the design.

So in terms of actually doing that work it was meant to be after receiving the design application. Of course we would need to prepare beforehand.

Like Don said we do put out a RIS once a year asking potential applicants or vendors when they think they might come in and we take that into account in our scheduling and resources. And we do plan at some point to be ready to address a technology-neutral framework review.

CHAIRMAN BLEY: Okay. But it's also in that extension to the SRP. You had a Rev 1 that talked about how you'd look at the small modular reactors. And it was like stage 3 of that process I think.

MR. MAYFIELD: This is Mike Mayfield from --

CHAIRMAN BLEY: Good to see you, Mike.

MR. MAYFIELD: -- Reactors. I like this column. Unfortunately Dr. Kress can see me all too well.

We did, when it looked like we were going to be challenged to see an application under NGNP we started looking for alternatives where we could try and test drive some of it, it being the technology-neutral framework.

And so we were going to try and use NuScale as the test case. Some vagaries with their schedule. They are -- that vendor is still committed to be part of it. However, vagaries in their submittal schedule coupled with current budget stressors make pursuing that at this time not tenable.

CHAIRMAN BLEY: Fair enough.

MR. MAYFIELD: We haven't zeroed things yet but it's getting dangerously close and likely that we will not be able to proceed with that anytime soon.

But that's -- going back to Dr. Corradini's question if we start getting serious indicators of a submittal we'll go back to the Commission and try and figure out how we could fund such a thing. It would not be a trivial budget impact.

MEMBER CORRADINI: So can I ask a kind of follow-up, Mike? So, some of the conclusions in the summary, in the summary report which kind of toss it back to the Commission from a policy standpoint. Is that one way of saying that things that are really going to be tough to address the Commission is going to come back with a policy decision? There are things relative -- now I've forgotten. There were three or four of them. And you really can't go forward with a technology-neutral framework until some of those are clarified.

MR. MAYFIELD: That's correct. And what we have talked about with some of the Commission assistants is we're not going to bring them something that's ill-defined and ask for a policy determination.

In fact, that's a conversation I'd had with Tom O'Connor when we started talking about what we could do with this assessment report was what of these could we dress up as policy determinations and put in front of the Commission at this time. And the dialogue with some of the Commission assistants was that's just not going to take them anyplace useful because they need to have something specific to address the policy on.

So this -- we felt like the effort that Don and the team are going to describe this afternoon was probably as far as we could push it at this stage, recognizing that there will be issues that we will want to take to the Commission for policy determination as we would move this forward based on a specific design application. And being able to move those forward could predate an application. We'll just have to see how this unfolds with a little time.

DR. KRESS: Will the TVA plans on small

modular plant fit in here anywhere?

MR. MAYFIELD: I'm sorry, please say it again?

DR. KRESS: Will the TVA plans on doing a small modular LWR fit into this anywhere? Or do you know? Maybe I'm bringing it up too soon.

MEMBER CORRADINI: So are you asking about mPower?

DR. KRESS: No, I'm talking about the TVA plans to use the Clinch River Breeder Reactor site.

MR. MAYFIELD: Well, TVA has told us that they intend to put the mPower, and in fact in their RIS response put mPower on the old Clinch River site.

DR. KRESS: Right.

MR. MAYFIELD: So they're moving forward with that. They have been less enamored -- well, BMW has been less enamored with being the pilot on this.

DR. KRESS: Yes, they're probably use the normal procedure I guess.

MR. MAYFIELD: Normal meaning Part 52 where B&We and TVA continues to look at Part 50.

DR. KRESS: Thank you.

DR. CARLSON: Okay, so I'll get back to my presentation. Again, we will finalize and issue these to DOE as enclosures to a letter publicly after ACRS review in May.

The presentations today are based on this first product, the issue summary report, but we will also dip into the more detailed white paper assessment reports for certain details.

Major conclusions, and we'll get into the details, but just very high-level. The staff used the proposed approaches to these licensing issues are generally reasonable with a number of caveats.

And at the high level the caveats are that deterministic elements of the RIPB approach should be strengthened. Really what that means is instead of what we thought should look like an option 2 we think some of it looks like option 3 in certain respects. And so this is the staff's advice on how to get it back more into option 2.

Another key element is consistent with the licensing strategy report to Congress. It called for licensing the NGNP within -- by adapting the current licensing framework as a prototype specifically under the prototype testing provisions of Title 10 Code of Federal Regulations 50.43(e)(2).

Our review of some of the technical issues under fuel qualification and mechanistic source terms reinforced that with specific issues that would have to be resolved by prototype testing. And we'll get into that.

MEMBER CORRADINI: Okay. So you're going to explain what all that just meant. Because early on in the licensing strategy when ACRS wrote the letter saying go forth, that looks like a good strategy, originally DOE had the option of using this as a demo. It wasn't 10 C.F.R. 50.43(e), it was another part where it would be a test reactor and the test reactor would have various power stages and they would only get an ascension from zero power to 5 percent, from 5 percent to 10 percent as they proved out. So how is (e) different than that? DOE chose not to do that and NRC agreed with that at that time. Is this different? This sounds very similar.

DR. CARLSON: It is.

MR. MAYFIELD: This is Mike Mayfield again. It is. That's essentially the prototype provision. So through licensing conditions on the design you would impose either additional trip set points, a power ascension program.

MEMBER CORRADINI: And that's proven empirically --

MR. MAYFIELD: As proven empirically. Then you would start to remove -- well, upon request you would start to remove those license conditions or modify them to allow power increase, to remove some more conservative trip set points, whatever conditions you might have imposed to assure safe operation while you were proving out the design.

MEMBER CORRADINI: Okay. And then you're going to explain it, but just at a high level. So DOE has seen this second conclusion. Have you seen any response from DOE? Because this has been discussed in the past, 10 years ago with DOE and this was not a path chosen. So I'm kind of curious on the back and forth in terms of the philosophy of this.

DR. CARLSON: We have had some discussion. I didn't detect that there was fundamental disagreement on this point.

MEMBER CORRADINI: Okay. Because just to take you back historically, in 2003 the INL's internal independent review team suggested this approach if I remember correctly in 2003.

DR. CARLSON: That said, when we do discuss these things and it's in public meeting records there's an indication that the NGNP industry alliance is hesitant to go to prototype depending on what you mean by prototype. A prototype can look a lot like the standard plant and that's what they want.

MEMBER CORRADINI: Okay, fine.

DR. CARLSON: Yes.

MEMBER CORRADINI: All right, thank you.

DR. CARLSON: So, important qualifiers. Of course this is pre-application review. It's for a design that's pre-conceptual at best. And we haven't seen any real analysis. So the staff feedback is advisory and represents no regulatory decisions and no final positions on any issue.

Our regulatory positions will be based on the NGNP license application and related Commission policy determinations that may be provided in the future.

And again there were comments today and back during the January 17th briefing about applying this as a technology-neutral framework. And indeed certain elements have been described by DOE as technology-neutral. The top-level regulatory criteria, the frequency consequence curve itself is, in principle that's all technology-neutral.

We did not look at it from a technology-neutral perspective. We looked at it solely as it would apply to a modular HTGR design concept. If we were to look at it for other technologies I'm not sure we'd reach the same conclusions.

So, what do we mean by modular HTGR design concept? I think DOE-INL has presented that very well.

I have been immersed in high-temperature gas-cooled reactor technology off and on, mostly on, since 1978 so you might guess that I have presented some overviews of what an HTGR, in particular what a modular HTGR is. And I've always found that there are one or two slides that make people say ah, that's what it's all about.

First of all, the early history of HTGRs. There was the first generation of HTGRs, the Dragon reactor in the UK 1966 to `75. Then in Germany the AVR 1967 to 1988. That's a picture of the AVR taken from the building where I worked from 1978 to `83. So I was part of that German R&D program, very intimately familiar with the AVR.

So it was a pebble bed reactor, 46 megawatts thermal. It was truly very high temperature, 950 degrees C outlet temperature for much of its operating life. And it had about 70 percent capacity back there as a test reactor.

Then there was Peach Bottom 1 in the United States in the same time frame, `67 to `74, a block type, 115 megawatts thermal, 725 degrees C outlet.

Then the second generation, Fort St. Vrain and THTR. THTR was at a high level a lot like Fort St. Vrain except the core was different, it was pebble bed, not prismatic.

So this is one of those pictures that make people say aha. I would like to present a version of this slide that shows the AVR, the German, because it's very similar. So you could replace Peach Bottom with the AVR. You could replace Fort St. Vrain with THTR and you could replace the large one there with PNP-3000. When I worked in Germany I did a lot of analysis on the PNP-3000.

Then TMI happened and the reaction of the HTGR community in the U.S. and Germany, and there was very close collaboration. The HTGR community consisted of General Atomics and mostly Oak Ridge at that time were having a lot of exchanges and interaction with the Germans. Mike Kania certainly can attest to that. He was part of that.

And their reaction to TMI was saying how can we make this really inherently safe. So this graph I think says a lot. You see on the ordinate the peak fuel temperature, the maximum fuel temperature in their accidents. So basically these are accidents where you depressurize the helium and it heats up gradually to some peak temperature.

So you see for Fort St. Vrain with large HTGRs you would get up to temperatures that would sublimate the graphite and melt the fuel if you did nothing. So they relied on active features to prevent that.

The mind shift that occurred post TMI, and it happened over a couple of years, and I was fortunate to witness it there in Germany was through a series of seminars. Ended up really in late 1981 with the seminal paper by Lohnert and Reutler, the advantages of modular. And so that was the birth of the modular concept and General Atomics was right on board. So in the early eighties we saw the emergence of what we now call the modular HTGR design concept.

And as you heard today it's lower power density, different core geometry, long, slender, passive conduction of decay heat through to the reactor vessel to a reactor cavity cooling system so that the peak temperature in the core and localized in the core is below a safe temperature, well below 2,000. Sixteen hundred has been a limit. Dr. Petti is suggesting that that limit may be 1,700 or higher depending on how their fuel program works.

So, before I move onto the next slide I'd also like to point out there were two factors that converged, TMI in 1979 and then in 1980-81 the Germans for the first time demonstrated really high-performing, high-quality TRISO fuel. So those two factors were crucial to coming up with this concept of modular HTGR safety.

So these are modular reactors. And sometimes they're called small modular reactors meaning low in power. But in terms of geometry the word "small" doesn't apply.

So this picture shows two PWR reactor vessels fitting neatly in the reactor vessel of a 600 megawatt thermal prismatic HTGR design. So a key point is that that's really what low power density means, big reactor.

Per-unit power, modular HTGRs are much larger than light water reactors. They have much, much lower power density, on the order of two orders of magnitude lower. They have much less fuel in the active core in terms of volume.

Light water reactors are 30 percent fuel, HTGRs in the active core are a half percent. If you improve the reflectors which have a lot to do with the thermal inertia it's less than 0.2 percent. So the big message here is they have tremendous thermal inertia. And so with that, that's the modular HTGR design concept we're talking about.

So, now we're going to start going through our feedback on the issues. And we're going to start with licensing basis event selection. And again we're doing it based on the contents of our issue summary report.

And we start with licensing basis event selection because that's the most obvious thing. If you're -- the licensing basis events that you use for light water reactors don't really apply to this technology and so you have to come up with a new set of licensing basis events and the option 2 framework is what we're trying to implement to do that.

So I'm going to now turn it over to Tom Boyle and Jonathan DeGange, first Tom who provided our feedback in response to specific requests for feedback in the July 6 DOE letter. And in so doing we're going to briefly paraphrase the requests and then provide our feedback in summary form. So I turn it over to Tom Boyle and he'll turn it over to Jonathan.

MR. BOYLE: My name is Thomas Boyle. I'm a project manager in the Division of Advanced Reactors and Rulemaking. And I'm going to begin with a brief review of the licensing basis event selection material that the staff reviewed when preparing these assessment documents.

All the material on these slides and the one following is in reference to the white paper submitted by DOE-INL. DOE-INL proposes a process for selecting, categorizing and evaluating licensing basis events that combines both risk-informed and deterministic outlooks.

This is meant to be consistent with option 2 of the 2008 licensing strategy report to Congress which indicates that deterministic engineering judgment and analysis should be complemented by NGNP design-specific PRA information.

Additionally, in the SRM to SECY-03-0047 the Commission approved the staff recommendation to allow the use of a probabilistic approach in the identification of events to be considered in the design provided there is sufficient understanding of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties.

The approach proposed by DOE-INL appears reasonably consistent with this guidance and would yield four risk-informed event categories, the anticipated events, design basis events, design basis accidents, and beyond design basis events.

MEMBER SKILLMAN: Tom, before you proceed, on your bullet number 1, please. Is there a clearly defined demarcation between where deterministic stops and probabilistic must begin?

MR. BOYLE: I'm not sure about that but we can -- when we can get into the staff evaluation of these different issues we can touch on some examples if that would help.

MEMBER SKILLMAN: We'll come back to it. Thank you.

MR. BOYLE: And this set of design basis accidents would be derived from DBEs assuming only safety-related SSCs are available to mitigate the consequences.

Offsite dose consequences of LBEs would be evaluated against the top-level regulatory criteria and EPA protective action guidelines. And the next slide will show this on their frequency consequence curve.

The SSCs would be classified according to their safety significance, safety-related and non-safety related. SSCs that are relied upon to perform safety functions that prevent or mitigate the consequences of DBEs or to prevent the frequency of certain BDBEs from entering the DBE range would be classified as safety-related. SSCs that are relied on to perform functions that prevent or mitigate the consequences from AEs or that prevent the frequency of certain DBEs from entering the AE range would be classified as non-safety related but would be subject to special treatment commensurate with their safety significance. All other SSCs would be classified as non-safety related and would not be subject to special treatment. Next slide.

MEMBER STETKAR: Tom, before you switch that I need to understand something fundamental because I've missed something somewhere. Could you tell me what an event sequence is?

MR. BOYLE: It's the entire plant response to an event. Not just the initiating event but all subsequent events to go along with that.

MEMBER STETKAR: Okay, I've got that. Could you tell me what an event sequence is in terms of the way it's used in this process? Let me give you an example because, you know, in the interest of time. The white paper and the staff's assessment of the white paper seems to bounce back and forth among the concepts of an event sequence family and an event sequence.

Now I will refer you to Figure 4 in the white paper which is a little picture cartoon of an event tree. The event tree has a couple of functions in it. One is shut down the reactor. The other is take heat away from the reactor.

The take heat away from the reactor function has three top events and they are listed in that event tree in a certain order with failure probabilities associated with each top event. Depending on the order of those top events you will have different frequency assignments to each end state which will give you different conclusions regarding whether a particular sequence in my connotation of a sequence, a path through the event tree, is either an AE, a DBE or a BDBE.

So that the definitions of top events and the sequence of those top events in this particular event tree, defining a path through the event tree, determine whether a particular sequence is assigned to a DBE, BDBE, or AE category. So I need to understand what an event sequence is.

MR. BOYLE: Well, when you --

MEMBER STETKAR: Because I understand that each sequence has supporting it, you know, tens of thousands or billions and billions of cut sets that all lead to that same failure of a top event. And in that sense each failure, top-event failure is the accumulation of several functionally identical cut sets. And some people will call a cut set a sequence. But in the context of the way the methodology is presented in the white paper it is presented along the lines of tracing a sequence through an event tree.

So I need to understand what that sequence means because if I develop a sequence by just taking those three heat removal top events and switching the order of them I get a much different characterization of AEs, DBEs and BDBEs. The BDBE is the same because it's failure of everything.

But an event sequence is a combination of successes and failures. And the intermediate success states could really result in different release categories, not the worst possible release category, but different intermediate categories which you then play against your frequency consequence curve. So it's important for me to understand how those successes and failures combine and have you thought much about that.

MR. BOYLE: I'm not sure I understand the problem here. So you're saying that a given --

MEMBER STETKAR: We can -- in the interest of time maybe at the break I'll show it to you graphically a bit. But the definition -- if a sequence is a path, a functional path through an event tree, combinations of successes and failures, as it's presented in this Figure 4 then the order of the top events and how you define a particular top event can change your conclusions.

MR. BOYLE: Right.

MEMBER STETKAR: So I'm curious how that process is going to be implemented in practice so that there's consistency from one design team doing one PRA for one particular design to a different design team doing a different PRA for their design. And I didn't see anything in your assessment paper that addressed that.

DR. CARLSON: There were some very high-level statements in the licensing --

MEMBER STETKAR: There are very high-level statements.

DR. CARLSON: Exactly, talking about event sequence families and that they would exercise SSCs in similar ways. And one that would be more challenging would be representative of that family.

MEMBER STETKAR: But there are also examples in the white paper, for example, where an event tree is developed partially and a failure branch in that event tree is not developed further because that failure branch drops below the magic 1E^{-8} frequency. And they say well, okay, there's a half a dozen other sequences out in here but because this drops below.

The problem is that some of those half a dozen other sequences actually add up to more than your problem. So there they're truncating on this failure path through an event tree, not families of cut sets. They're truncating actually on this cartoon figure path through the tree which doesn't seem to be consistent with the notion of event sequence families. It seems to be a literal interpretation of a flow path through the tree.

DR. CARLSON: I think we understand your comment in general and truly we understand that need to fully develop this concept.

MEMBER STETKAR: You can do it in practice but I was surprised that I didn't see more discussion of it in the exchanges.

DR. CARLSON: It was presumed at a high level and so we didn't get beyond the high level.

MEMBER STETKAR: I get it at the high level. I honestly get it at the high level provided that this accumulation process, this development of whatever you call an event sequence family, whether it's a bucket of cut sets or whether it's similar paths through an event tree is done according to the high-level discussions that I can read.

But a lot of the specific examples that I see seem contrary to that notion, or at least not fully consistent with that notion let's say.

DR. CARLSON: Yes, we agree. There's a need for a lot of specificity that we really didn't get to in this process.

MEMBER STETKAR: Okay.

DR. CARLSON: We kept it at a high level. But what is the event sequence family, how is it defined, how is it treated. We don't have --

MEMBER STETKAR: Okay. I've gotten a little bit of my answer back so thanks. I appreciate that.

CHAIRMAN BLEY: The paper, your paper doesn't quite warn -- because you wrote the white paper of concern in this area.

DR. CARLSON: It's a good point. No, it doesn't. I think that's true.

MR. BOYLE: Anything else about this slide?

CHAIRMAN BLEY: Not yet.

MR. BOYLE: Sounds good. Let's move onto the next slide then.

I'll briefly point at some of these event categories and everything on this just to kind of rehash. And note that these event frequencies are shown per plant year rather than per reactor year. In the DOE-INL proposal event frequency cutoffs are independent of the number of modules. But as a plant consisting of 10 modules would have the same event frequency cutoffs as a plant consisting of 1 module.

At the top we have the anticipated events that are expected to occur within the lifetime of the plant. They're expected to be more than 1 times 10^{-2} per plant year. The basis for the dose consequence criteria for AEs is 10 C.F.R. Part 20. The reference value is 100 mrem TEDE cumulative annual dose, the EAB, mechanistically modeled, realistically calculated.

Design basis events expected to maybe occur within the lifetime of a fleet of plants range from 1 times 10^{-2} to 1 times 10^{-4} per plant year. And the design basis accidents as we said before are derived from the DBEs by assuming only safety-related equipment responds.

The basis for the dose consequences for both DBEs and DBAs is 10 C.F.R. 50.34. The reference value is 25 rem TEDE at the EAB mechanistically modeled and conservatively calculated.

Beyond design basis events are off-normal events of lower frequency than DBE. They are evaluated to ensure they do not pose an unacceptable risk to the public. Frequency is greater than 5 times 10^{-7} and the dose circumstance for these DBEs are based on NRC's QHOs, mechanistically --

DR. KRESS: Did I understand you correctly, that 5 times 10^{-7} is on module basis rather than plant basis?

MR. BOYLE: That's per plant year. It's per plant year.

DR. KRESS: It's per plant.

MR. BOYLE: That's correct.

DR. KRESS: I misunderstood.

DR. CARLSON: Always on the ordinate for plant year. Since the late eighties.

DR. KRESS: I misunderstood what he said about the cutoff frequency.

MR. BOYLE: Let's go to the next slide.

MEMBER STETKAR: Tom, is it appropriate to ask my question about why the apparent inconsistency in the use of uncertainties on the consequence scale is applied in this framework? At this time or do you want to address it later?

MR. BOYLE: I believe Mr. DeGange will get to field that question.

MEMBER STETKAR: Okay. That's fair.

(Laughter.)

DR. KRESS: How about my comment on the fact that those stair steps give some inconsistencies. And that it probably would have been better to have a straight line top-level regulatory criteria because you get rid of those little ambiguities about when you cross over one spot and another.

DR. CARLSON: We did have an RAI on that and there was a good response to that. And where the staff said we're not proposing an FC curve but what if and we came up with the straight line curve and we discussed that a little.

DR. KRESS: Okay.

DR. CARLSON: And we have a backup slide on that too if you want to get into it a little more.

MEMBER CORRADINI: With various colors no doubt.

DR. CARLSON: Of course.

MR. BOYLE: In general the staff feels

that the DOE-INL approach is reasonable. It appears to be too risk-based in some places.

The LBE selection process should incorporate more deterministic elements as described in the coming slides. And we have a selection of licensing policy technical issues related to LBE selection that the staff has identified during its review such as the frequency cutoffs for DBEs and BDBEs, the per-plant year method of assessing multi-reactor module frequencies, processing criteria used for selection of DBAs and alternate TLRC and FC curves for future HTGRs or technology-neutral frameworks.

DR. KRESS: Frequency cutoff is just to be sure you're below the fatality QHO.

MR. BOYLE: Okay.

CHAIRMAN BLEY: I'm a little curious as to why they're per-plant year approach requires Commission policy decision. It's -- I don't want to use the more conservative, it's more realistic than what we're currently doing. Doing it for them doesn't say you have to do it for everybody but it's certainly something some of us have long thought ought to be applied on a particular site. So I'm just curious why -- what they're doing which is really restricting themselves to meet the rules for the whole site requires a policy decision.

MR. BOYLE: I believe it was an unresolved issue in a SECY paper that never got a response. I believe Mr. DeGange will talk about that a little bit too.

CHAIRMAN BLEY: He's stuck with everything.

(Laughter.)

DR. KRESS: The reason for my question about the prompt fatality and safety really being the reason for the cutoff value is that worries me because there is such a thing as societal risk, total number of deaths, land contamination, cost of all that stuff which probably controls the where you're cutting off the frequency on the basis of prompt fatality safety goal. Are you really going to evaluate the total societal risk for beyond design basis accidents?

MR. BOYLE: I'm not sure we're cutting off the frequency based on when the consequences at the QHO. They're saying --

DR. KRESS: You'll go ahead and do the level 3 no matter what the frequency is.

MR. DEGANGE: I think that what this bullet's trying to get at is the frequency that would be used for categorizing is it going to be in the design basis event region or beyond design basis event region. Not looking at the actual QHO.

DR. KRESS: Well, okay, but that's the reason -- if it meets the top-level regulatory criteria. Pretty darn sure it meets the QHOs. Only if the QHOs and the two you've got then, then I'm questioning whether or not those are the right place to look at. And maybe you ought to think about that.

DR. CARLSON: Yes, I think the staff is thinking about having criteria for --

DR. KRESS: Come up with another QHO maybe? A new one?

DR. CARLSON: I don't think that there's anything pending but it's certainly just being discussed.

DR. KRESS: As long as you're aware of it.

DR. CARLSON: Yes.

MR. BOYLE: With that we go to the first issue. This first issue is the DOE-INL's event categories and proposed descriptions. Again those categories are anticipated events, design basis events, design basis accidents and beyond design basis events.

These event categories and descriptions appear generally reasonable. However, the staff feels that the full selection of LBEs should include more deterministic elements.

For example, the definition of DBA proposed in DOE-INL's LBE white paper is different from the one traditionally used by the staff. To be more consistent with current regulatory practice and to be more in keeping with option 2 in the licensing strategy report the staff feels that the full set of DBAs should include event sequences populated by the applicant and/or the staff even if those events would otherwise fall within the BDBE frequency range or below.

MEMBER STETKAR: Why is that, Tom? I mean if you -- if this is a reasonable regulatory framework and if the PRA is developed to comprehensively evaluate the whole spectrum of internal events, external events, any hazard to the plant, why require a separate special evaluation of, as you characterize it, postulated deterministic event sequences, or I've seen them listed as hypothetical event sequences. Because the PRA should already have evaluated those event sequences.

MR. BOYLE: If that's the case then we won't have any additional postulated events. If the PRA really does cover everything that the staff feels would be --

MEMBER STETKAR: Okay, I didn't get that from your assessment paper because --

CHAIRMAN BLEY: Neither did it.

MEMBER STETKAR: -- my interpretation of the PRA as has been characterized in the DOE white paper is it is a full-scope, comprehensive, all internal/external hazards, all modes of operation PRA. That it's comprehensive. And I thought you were interpreting it that way and still saying yes, but even though they've quantified an event sequence at 10^{-100}, pick a number, I'm going to require them to evaluate this because I think it's an important event.

DR. CARLSON: I can take a stab at that. You want to go first?

MR. BOYLE: I'll just say that could be. It's possible that the staff would want to see that 10^{-100} event. It's unlikely something that ridiculous. But it's --

MEMBER STETKAR: But my question is why. Why. Because if you adopt this notion that the risk assessment, and it addresses uncertainties so you have both horizontal and vertical uncertainties quantified.

MR. BOYLE: I'm thinking it's just too drastic a departure from what we're doing now. Like we're switching gears very fast going from basically purely deterministic to this is entirely almost risk-based.

MEMBER STETKAR: But --

MEMBER CORRADINI: Can I ask the question differently? Just to take the gas reactor out of it. If you had their backup slide and you essentially mapped onto it the light water reactor instead of a full-scope PRA are you trying to tell me that the design basis accidents that you're requiring of current reactors don't exist in that population of little circles with bars and you've picked something that's stylized enough that it doesn't appear in the PRA? I don't think so. I'm looking at these guys since I haven't had one.

CHAIRMAN BLEY: It would be an indication that there was something wrong with the PRA.

MEMBER CORRADINI: Well, it infers a lack of completeness. So I can understand that, that there's always a lack of completeness. But on the other hand I'm trying to figure out that if I took away this technology -- I'm sorry I'm driving you back to technology-neutral but it seems the logical thing -- and you put back in the light water reactor which is when we started arguing about 1860 6 years ago, I think you were still on the committee.

DR. KRESS: Yes, I was.

MEMBER CORRADINI: You almost do a test drive on light water and ask the question how I get the DBAs from the LBEs. And their approach at least assuming they do all of this in a relatively complete manner would be essentially what you do for light water reactor. Otherwise you have to say everything in the PRA for light water reactor is incomplete and so we're going to invent one over here just to make sure we've captured it. That strikes me as odd.

DR. CARLSON: I would like to take a stab at answering the question why and I think it will help the other questions as well.

The option 2 was selected for this technology and not option 3. And I think a big reason, a big thought process, a major thought process behind selecting option 2 was we don't have a lot of operating experience with this technology, we have -- there has never been a modular HTGR design built or operated. So it's going to be difficult to assess the reliability of the PRA information. There's going to be more than the usual amount of subjectivity in determining what are the uncertainties, et cetera. Whereas for light water reactors there's a considerable database of being able to say yes, we understand the reliability of PRA information pretty much for that technology.

So perhaps if we were thinking about a licensing approach and change out modular HTGR for some light water reactor design maybe we would say something more like option 3.

MEMBER CORRADINI: Can I say it back to you, what you just said? What you're telling me is that not only might the PRA be immature but even the uncertainties in the PRA would be immature. So you wouldn't necessarily take the upper right-hand corner of all their uncertainties. You'd have to add something to it because you're not sure. That's what I hear you just saying.

DR. CARLSON: One of the recommendations that we'll talk about further, we mentioned it in our documents, is it's important to have a peer review of the PRA. And I can just imagine the peer review having a lot of diverse opinions about how well you characterized uncertainties.

CHAIRMAN BLEY: I think in this entire framework, now we're just talking about licensing basis event selection. But you also have the defense-in-depth side which is framed to allow you -- them to as they do their PRA consider areas where they might not have a good basis for evaluating the uncertainty and that would call for some deterministic defense-in-depth support.

I'm not sure why it belongs in here. If it's the kind of thing that Tom was mentioning there's a scenario they didn't consider. Well, when you review the PRA it ought to be added in. If there's a real scenario. The scenarios we usually use for PWRs and BWRs might not be appropriate here. So that's not the place to go. So it seems to me there's a place already in this structure to account for that rather than saying well I had some deterministic events here.

And I agree with these two guys that we'll add new events seems really surprising. And if there are some that aren't in the PRA that ought to be added. But this issue of maybe there are uncertainties here that we don't fully understand might call for additional defense-in-depth to protect us.

DR. CARLSON: That is the value of course being risk-informed. It helps to identify sequences that you wouldn't identify using pure engineering judgment deterministically. And so that's the value.

But, okay, how much can you rely on the

risk information and how much do you need to compensate for uncertainties with engineering judgment.

CHAIRMAN BLEY: Of course that works both ways. The PRA really would have brought engineering judgment in and expanded what you might have had if you hadn't done that full structure as well.

DR. CARLSON: So we definitely would consider events that are identified from the PRA. And so their LBE approach that is built around PRA would be a source of licensing basis events including design basis events, design basis accidents. But we think that to cover uncertainties we would need to postulate some deterministic events. And we'll have --

CHAIRMAN BLEY: I guess I still fall back -- if you come up with new event sequences, new initiators, new event sequences, you shouldn't just postulate them as DBAs. They ought to go back in the PRA to get fully evaluated.

MEMBER RAY: What do you mean fully evaluated?

CHAIRMAN BLEY: Evaluated probabilistically. As part of the PRA. The complete PRA.

MEMBER RAY: -- stage is that practical?

I mean it seems to me like it's almost a semantic discussion.

Take for example the blowdown of the helium system. Has that got a potential to interact with the passive cooling system in such a way as to disable it? Well, God knows.

CHAIRMAN BLEY: Well, we can -- I suspect we can do a lot better than God knows on that question.

MEMBER RAY: Well, I don't know. At this stage is what I'm asking.

CHAIRMAN BLEY: Oh, at this stage. Of course not. At the stage we have a real design -- this only comes up when you have a real design using that. This isn't being done now. This is just how we'll do it when we get there.

MEMBER CORRADINI: But I guess just to make sure at least where I'm coming from. I think we're all asking kind of the same question. It's not that I wouldn't disagree with your judgment that things are not at the appropriate stage of maturity so you're going to have to add some level of we'll call it engineering judgment on top of it. That I don't mind.

It's just the way it's characterized I

would expect the reasoning is that I'm uncertain in this direction, I'm uncertain in this direction. They probably have identified some sort of set of sequences that I'm worried about. But it might be over here and it may even be more than over here because they don't even understand the uncertainties. So I'm going to add adjustment.

But that's not the same thing as saying I'm going to come up with a stylized thing that out of the blue, blink, it's over here. Although it strikes me it's much more you've got to work within the context of what they're already identifying. And then you say well, because of some physical process that we're unsure of we're going to add some judgment, some wiggle room. That I can understand.

DR. KRESS: Let me ask a maybe related question for the LWRs that determine safety categories and SSCs by using importance measures. These are importance measures on the core damage frequency usually. I don't see -- I'm not sure that this process we're looking at to see if it meets the top-level regulatory criteria is actually equivalent to that.

DR. CARLSON: You're right. There is no level 1 PRA for this technology that we can make sense of. There is no CDF in this technology. A core damage state that relates to the type of core damage --

DR. KRESS: I hate to admit to that but you could have an FC curve that is equivalent to the core damage frequency and a LERF together. I don't know if you've thought about it but you could have one.

DR. CARLSON: For this technology you're talking about.

DR. KRESS: Oh yes, you can for this technology. It's an FC curve.

MEMBER CORRADINI: But Tom, can I ask Tom a question? If we go back to what Dave presented relative to his testing at high temperatures you're looking for a release of radionuclides at some temperature. That's no different than a degraded state. You can have an intermediate analysis based on --

DR. KRESS: Exactly what I was thinking, yes.

DR. CARLSON: We have talked about it will be other criteria that you use in reviewing a licensing basis event that engineers use. We talked later about equivalent to a specified acceptable fuel design limits for this technology hasn't been developed yet.

DR. KRESS: Well, talk to you about this compiling the original deterministic approach I'll call it to this new technology-neutral approach. If you did that on just looking at the safety significance of SSCs I don't think you'd get the same integers. That's what bothers me. I think you'd get different categories, different SSCs classified as safety compared with the old process which just uses importance measures compared to what we're doing now. I'm not sure because I haven't done any myself but it just appears to me like you would. But it's just a question.

DR. CARLSON: Well, we'll take it as a comment now and we'll try to address it later.

MEMBER STETKAR: I think a little bit of my concern is this notion of what is a postulated deterministic event sequence. Okay, I can postulate a deterministic event sequence that says you must assume that you have full core damage. That's a postulated deterministic event sequence. Now, you must protect the public from that.

DR. CARLSON: I don't think that that was what we proposed to the Commission in past SECY papers on this topic. It was very event sequence-specific mechanistic. And so you're postulating --

MEMBER STETKAR: And if that's the case then I fall back to what Dennis and Mike were saying is that identifies a deficiency in the PRA which ought to be resolved through the PRA process, that sequence that you've identified.

MEMBER CORRADINI: In the meantime though you would identify it's something you've got to consider.

MEMBER STETKAR: Yes. But it's something that needs to be considered in the context of the risk assessment with its evaluation of the consequences, the uncertainty, the frequency and its allocation among the three nominal categories of events. It doesn't automatically become a design basis event requiring special attention simply because you've identified it.

CHAIRMAN BLEY: But once it's in there then if you're still not comfortable with the way the uncertainties have been characterized and think they might be greater than certainly defense-in-depth calls for doing something more I suppose.

DR. CARLSON: So the balanced approach of risk-informed and deterministic methods, basically the policy is use risk information and use deterministic engineering judgment to bound uncertainties. So I think that's the spirit of our overall recommendation that you need. Hard to say what the uncertainties are in the PRA for this technology because there's just not a lot of -- you have to get them operating to see what surprises there are.

DR. KRESS: I wasn't sure whether or not the air ingress accident was part of the PRA. It sounded like one of these things you're talking. The air ingress.

DR. CARLSON: We'll talk about that during my talk after the break. Okay.

MR. BOYLE: Well, we were also -- there's also the issue of the SAFDLs. The staff noted that INL stated that it plans to develop specified acceptable fuel design limits, or SAFDLs, for the HTGR fuel since the SAFDL structure that's been established for LWRs is not applicable to HTGR fuel. The staff would expect any SAFDLs for modular HTGR to ensure substantial margin to dose limits in the AE region.

DR. CARLSON: If you want a surrogate this might be the one you're after.

MR. BOYLE: This next issue is related to SSC classification. The staff believes that the DOE-

INL approach, the safety classification of SSCs, is a reasonable one. Staff believes that this approach is consistent with NRC's policy statement on PRA and expects that the applicant's selection of safety-related SSCs will comply with the regulations at 50.2.

The SSC classifications also appeared to reasonably address applicable traditional AOOs or anticipated operational occurrences. For AOO type events that fall within the AE region SSCs classified as non-safety related with special treatment would be available to prevent and mitigate the consequences.

Should one or more of these non-safety related with special treatment SSCs fail to respond to the event then the event might now fall within the DBE region where there would be safety-related SSCs to prevent and mitigate the consequences.

The special treatments for safety-related and non-safety related with special treatment SSCs would be in accordance with the safety significance of the functions performed by that SSC. Specific special treatments would likely be determined when more design information or when the application is received.

MEMBER SKILLMAN: Let's go to my question from 45 minutes ago on your first bullet. Where do you see in practice this demarcation between where your deterministic method seems to give confidence and where you need to begin to invoke probabilistic thinking?

MR. BOYLE: I'm going to defer that to Don to be on the safe side.

DR. CARLSON: I hope this is helpful but when you look at what we traditionally do. And again we're adapting the traditional framework, adding some more insights of course at the option 2 but not option 3 level. But the traditional framework has design basis accidents like large break LOCA. We have rod ejection events. And so put frequency numbers on that and I don't think they'll necessarily fall in what they're calling the DBE range.

But we think that that's a reasonable way to go, especially for the prototype. Until we get some actual experience with the technology, see if there are any surprises when they actually build the prototype. Some uncertainties will be reduced by a few years of operation on the prototype with of course testing and surveillance, et cetera, to meet the requirements of 10 C.F.R. 50.34(e). Does that help?

MEMBER SKILLMAN: That does. Thank you.

MEMBER STETKAR: You're not going to gain a lot of experience for large break LOCAs, rod ejection events.

(Laughter.)

MEMBER STETKAR: Well, one would hope not. So you'll still face that issue hundreds of years in the future.

CHAIRMAN BLEY: Hundreds of plant years.

MEMBER STETKAR: Hundreds of plant years. I'm sorry. Reactor years.

DR. CARLSON: Some of the uncertainties are associated more with normal conditions or potential anomalies in normal operating conditions. So you can --

MEMBER STETKAR: Those you'll get.

DR. CARLSON: Yes.

MR. BOYLE: That's all. I'll hand the reins off to Mr. DeGange.

MR. DEGANGE: Hi, I'm Jonathan DeGange. And alongside Don and Tom I also work in NRO in the Division of Advanced Reactors and Rulemaking. This next slide is covering the third issue pertaining to licensing basis event selection regarding NRC agreement with proposed placement of top-level regulatory criteria on a frequency consequence curve.

The overall staff view on this is that the approach is reasonable. The TLRC as you'll see them in the FC curve that was shown a moment ago proposed to be used with the FC curve to establish limits on frequencies of event sequences and their associated radiological consequences. And then they would be used in the categorization and evaluation of licensing basis events and ultimately in categorizing the treatment of SSCs.

Staff feels that this approach is consistent with the approved recommendation found in SECY-03-0047 regarding issue 4 in that it places greater emphasis on the use of risk information to be considered in the licensing approach by allowing a probabilistic approach in the identification of events.

So as Tom had previously discussed one point about the FC curve in noting the top-level regulatory criteria are effectively looking at dose. In addition to addressing dose consequences in their associated TLRC the staff does believe that DOE-INL should pursue an appropriate regulatory limit to ensure the required level of integrity of the fuel barrier to assure safe operation.

DOE-INL has acknowledged this and the need for the development of these limits in both our interactions with them over the past few years and in a 2011 NGNP project status report.

DR. KRESS: Does that mean limit on the number of failed particles that exist in a given pure load? You know, the full source term comes from -- there's some particles that aren't -- that are failed before you load them in. And they're talking about a quality assurance program to control that. And I presume there must be a limit on it. Is that the one that you're talking about?

MR. DEGANGE: I think we were talking more about in the relation to SAFDLs or something along the lines of that. You're looking at operating temperatures and --

DR. KRESS: I see in case of temperature and the radiation and accumulation might fail particles. You're talking about limit on that part.

MR. DEGANGE: Right. Yes, sir.

DR. KRESS: Okay.

MEMBER CORRADINI: Wouldn't it be both? I mean I guess I was looking for an example. You read it. I looked at it. And I'm still a bit cloudy. So I thought there would be -- in some sense there would be some manufacturing QA --

DR. KRESS: Quality --

MEMBER CORRADINI: -- requirement --

DR. KRESS: Yes, that's what I thought.

MEMBER CORRADINI: And then some sort of temperature limit just in case. I'm waiting for the DOE to come up and say something. I'm just, I'm trying to spur some --

DR. CARLSON: We will certainly get to that. We talk about the fuel performance under containment functional -- functional containment.

MEMBER CORRADINI: Okay.

DR. CARLSON: And so I'll be talking about that. And so we can continue this discussion during that presentation after the break.

MEMBER CORRADINI: I'm just trying to understand what the bullet -- an example of the bullet. I can think of two and I wanted to mention them to see if I could get the DOE staff and their contractors engaged.

DR. CARLSON: I don't know, it's up to the --

MR. DEGANGE: I think maximum temperature.

MR. PETTI: Do you want me to say something, Don? Do you want the DOE perspective?

CHAIRMAN BLEY: I'd almost rather have that separately from this. The way we came in.

MR. DEGANGE: So yes. So finally one last thing is because we have never actually licensed a reactor with this approach the staff does feel that future Commission consideration may be appropriate for determination of using the TLRC as dose acceptance criteria for the event categories. Go to the next slide, please.

This next slide on issue 4 is covering the fourth issue pertaining to licensing basis event selection which is NRC established frequency ranges based on mean event sequence frequency. And overall again the staff finds that the proposed approach by DOE-INL for categorizing each event sequence based on mean frequency to be reasonable.

In the approach the mean frequency would be used to categorize an event sequence as an AE, DBE, or BDBE based on where the mean frequency falls in relation to the respective event category frequency ranges.

And as several have indicated earlier, I think Fred did in his presentation, in the event that -- when they're looking at uncertainties it would be comparing both the upper and lower bounds of the frequency uncertainty distributions. They would be looking both at mean frequency and mean consequence of event sequences.

And so in the event that you had events that straddled multiple regions the proposal would be to compare against the dose criteria for both of those regions.

Overall the staff has no issue with this approach and in light of the remarks from the previous slide about the licensing basis we find the approach reasonable. The staff views the approach to be reasonable.

DR. KRESS: So the straddle you're talking about would be 95 percent confidence level?

MR. DEGANGE: If I'm not mistaken that is the proposed approach. So they would look at 95 percent confidence on the entire distribution and then ultimately you'd be looking at the mean to determine in the event that --

DR. KRESS: You say you'd go ahead and put it into the other category.

MR. DEGANGE: Right. Where these can be compared against both categories. And so the one that was more restricted.

DR. KRESS: Well, that's a defense in depth concept.

MEMBER STETKAR: On the vertical scale, yes.

DR. KRESS: Well, yes, that's where I'm stuck.

MEMBER STETKAR: Yes, it is. Now do I get a chance to ask you?

MR. DEGANGE: Oh, I don't know. I might have to defer back to Tom.

(Laughter.)

MEMBER STETKAR: And then they postpone you. That's okay.

CHAIRMAN BLEY: I think we're at the point, yes.

MEMBER STETKAR: Should I ask?

CHAIRMAN BLEY: Yes.

MEMBER STETKAR: Thanks. And I understood that. You know, I understood because people traditionally have thought about uncertainties and frequency. Everybody's grown up by modeling pumps and pipes and valves and their frequency of core damage and all of that kind of stuff.

In this framework there is also uncertainty in the consequences. And I'll come back to my question of why in the DBE range and only that range do I compare the 95th percentile of the consequences with the FC curve but not in the AE region or the BDBE region.

CHAIRMAN BLEY: Where do you use the mean?

MEMBER STETKAR: Where I only use the mean. Because, for example, if I have a 30 percent probability depending on my uncertainty distribution if there's a 30 percent probability that I exceed the acceptance criteria for a BDBE event I'm okay as long as the mean is below it. Even though there's a 30 percent probability I don't need to necessarily consider any other additional defense-in-depth measures to reduce that uncertainty or to actually reduce the consequences if I think of it in an absolute sense. And I don't understand that. I honestly don't understand why.

DR. CARLSON: I think Jim who's going to present later has some --

MEMBER STETKAR: If he's going to do it later -- that's fine, I'll wait.

DR. CARLSON: He can help us respond to that now if he'd like.

MR. SHEA: We can do it now or later. Or both.

MEMBER STETKAR: Why don't we do it now because we keep pushing me off.

MR. SHEA: One of the concepts is --

MEMBER STETKAR: You have to identify

yourself.

MR. SHEA: Oh, I'm sorry. This is Jim Shea of the staff. One of the things you want to think about is this frequency consequence curve is really you're looking at the mean values of all these frequencies. And then you then apply for the various regions what makes sense.

And for example, DBE range, I mean I should say the AE range you're talking about actual plant operation conditions which in real life you measure the actual consequence. And so what you want to do is compare that against a best estimate or actual consequence type analysis.

And one could say for the BDBE range it's a similar concept is when you're advising your emergency -- I've got to make sure I'm saying this right -- emergency preparedness program for your procedures to get your, for example, your best estimate result of how an accident would progress you would want to use more realistic evaluation versus in the DBE range which actually are there to influence the DBAs or even the BDBEs would influence what your DBA is.

Once you've selected your DBAs out of this frequency consequence curve, whether it's through PRA or some deterministic selections those DBAs against a top-level regulatory criteria, siting, specifically, would be done on a very conservative analysis, a 95 percent type analysis. Does that make sense?

MEMBER STETKAR: No.

(Laughter.)

MEMBER STETKAR: It makes perfect sense to me in the DBE range. It makes perfect sense. I understand. I understand it. You know, and it is as Tom's mentioned a way -- you can think of it in terms of defense-in-depth. You can think of it in terms of confidence in your margins to whatever regulatory acceptance criteria you've set by whatever shape of that frequency consequence curve.

I still don't understand why the same concept does not apply in those two other regions. And especially if I'm using this whole process to inform the design and the licensing of the plant.

MEMBER CORRADINI: You're worried about inconsistency being developed as you cross across those lines. That's what I thought you were getting --

MEMBER STETKAR: I'm worried about -- yes. I mean I'm worried about why in the top region I don't look at the 95th percentile of the consequence and see where am I relative to that curve. And I don't -- I also don't down in the bottom region on the very severe accidents.

DR. CARLSON: I'm going to tag-team with Jim on this. Jim started to explain that. I would like to just amplify with regard to the AEs that that 10 C.F.R. 20 limit is not an event-based limit. That's an annual cumulative limit. Moreover it is monitored.

MEMBER STETKAR: I understand that. And if indeed my monitoring program suddenly determines that 7 years into the life of the plant I've exceeded that because I hadn't thought about something that I should have thought about when I designed the plant, then everybody has a problem. The regulator has a problem, the designer has a problem and the operator has a problem. So why not think about the possibility that those things might occur when I'm designing and building the plant.

There might be uncertainties in those annual cumulative releases that you've not thought about simply by looking at a mean value estimate for what you guess is a mean value.

DR. CARLSON: So you believe with -- by using the mean values you're creating a likelihood or a possibility that you would get into this situation where the monitoring will show that you're outside the limits.

MEMBER STETKAR: If you talked to financial planners back in 2006 on a mean value basis everybody was making an awful lot of money.

DR. CARLSON: Okay. So yes, Jim.

MR. SHEA: And when people violated the law using -- exceeding those values they went to jail. So it's almost the same concept. If you're designing your plant based on a best estimate in the AE range and then however you find in actual operations you're exceeding your limits then, you know, that's why we have those limits there. So the concept is really, you know, you can think of it as a design concept.

But the other aspect of it is in that AE range, and we struggled with this for a while till we kind of -- the light bulb went on and that is even in those range to protect the fuel limits which haven't been described yet through the design. But when the design comes in there will be some sort of fuel limits. We've alluded to them.

But at that point those fuel limits, and we fed back into our paper that those would also be done on some conservative basis. Maybe not necessarily 95 percent confidence but on a -- some sort of conservative level so that you have some -- you have some margin to your limits on your fuel.

MR. DEGANGE: The staff has talked on this. I was going to save this and talk about this a little bit later on. They're all kind of related.

In SECY-05-0006 the staff kind of hit on this point. And that did get brought up in the writeup that we did I believe. But we discussed the use of scenario-specific source terms for licensing basis there. And there was some issue of -- was brought up about using conservative calculations for -- versus best estimate.

MEMBER STETKAR: Okay. You know, in the interest of time, Dennis, I think I'll -- we've discussed this probably enough. I'll go look up that SECY if that's got a little more information in it. Thanks.

CHAIRMAN BLEY: And the committee will probably talk some about this offline.

MR. DEGANGE: So this slide is covering a request from DOE-INL regarding endorsement from the NRC staff on their proposed per plant year method for addressing risk at multi-reactor module plant sites.

So in the approach to account for multi-module plants the approach proposes expressing the frequencies of licensing basis events in units of events per plant year where a plant is defined as a collection of reactor modules that have selected shared systems.

Overall the staff feels that the proposed per plant year method being called upon is reasonable and takes no issue in its assessment. The staff believes that an integrated risk approach is ultimately more conservative and comprehensive than the treatment of modules on an individual basis.

DR. KRESS: Instead of the word "reasonable" I would have said "necessary."

(Laughter.)

MR. DEGANGE: It would enable the risk assessment to include event sequences that involve source terms from one reactor module or multiple reactor modules. So the staff overall finds it reasonable. The staff does believe that future Commission direction may be appropriate for this topic.

CHAIRMAN BLEY: I guess I'm still -- to do this I don't see why you need anything special. But if the Commission speaks on this then it may have implications for everybody else out there.

MEMBER STETKAR: I mean in some sense the Commission already has though in terms of the post-Fukushima issue of looking at multi-unit site accidents for conventional operating LWRs.

MR. DEGANGE: Back in SECY-03-0047 there were a number of different issues that were brought to the Commission to vote on. And one of those was consideration of a per plant year approach. And that was denied. And the reason being was that we needed to go to the ACRS first on that. And we did go to the ACRS with the issue. And what happened really in the letter back was there was a pretty mixed review of -- there were opinions both ways on the topic. And there was really never a definitive line of thinking that came out.

CHAIRMAN BLEY: My memory which could be really faulty on this was back then the request for a position from the Commission was really aimed at applying it to all sites for -- rather than just new modular reactors. But I'm not sure of that. Don, do you remember?

DR. CARLSON: I'll have to refresh my memory as well.

CHAIRMAN BLEY: Anyway, you don't have to go further on this. That's all record we can work from.

MR. DEGANGE: Okay. So the next slide, issue 6, is covering a request from DOE-INL regarding agreement on frequency cutoffs being established for the design basis event and beyond design basis event regions of the proposed FC curve.

They provide justification for these frequency cutoffs in their RIPB white papers that we have looked at thoroughly. I'd like to note that in the assessment we have done we have, as Don pointed out, are looking at their usage in the context of modular HTGR licensing and have not been assessed in terms of a technology-neutral context.

So as seen on the FC curve the proposed cutoffs for the beyond design basis event sequence frequency would be between 10^{-4} and 5 times 10^{-7} per plant year. And the QHO of the prompt fatality safety goal seen in NUREG-0880 limits the increase in an individual's annual risk of accidental death to a tenth of a percent of 10^{-4} per year which sizes out to an incremental increase of 5 times 10^{-7} per year.

So consistent with this QHO the NRC staff views that the lower-frequency cutoff for the beyond design basis event region of 5 times 10^{-7} per plant year is reasonable.

Regarding the DBE region with frequencies between 10^{-2} and 10^{-4} per plant year the staff believes that the lower cutoff of 10^{-4} is reasonable as long as the PRA used in the LBE selection process assessed multiple failures from common cause events and as long as it accounts for both operating in shutdown modes as well as internal and external plant hazards.

So like some of the other issues that the staff has talked about we think that future Commission direction may be appropriate for deciding the actual cutoff values when it comes time to actual licensing. So the next slide, please.

The last issue here on LBEs is covering a request from DOE-INL for an endorsement of the overall process for performing assessments against the TLRC addressing specific issues with uncertainties, calculational methodologies and the adequate incorporation of determinism.

So as previously discussed the staff feels that the approach overall their use of engineering judgment to address uncertainties is a reasonable approach for assessing licensing basis events in a risk-informed manner.

And additionally we think that the calculational methodologies proposed to be employed, they assess full event sequences using best estimate models with either mean or conservative analysis.

One potential point to point out that I think we've talked about just recently here. The staff position in SECY-05-0006 which discusses the use of scenario-specific source terms for licensing decisions.

In that SECY there is a discussion about using source terms for compliance and the usage of conservative values based on best estimate calculations. And this is consistent with DOE-INL's proposal for design basis events and design basis accidents.

However, for the anticipated events and beyond design basis event regions compliance with the top-level regulatory criteria the staff views the proposed approach of realistic source term calculations as meritable but would need further consideration. And notes that it would probably involve new regulatory interpretations likely to require consideration by the Commission.

Another thing to point out was the, as I think Tom mentioned earlier was on the topic of design basis accidents. DOE-INL proposes as you all are well aware that design basis accidents would be derived from design basis events with only safety-related SSCs responding and being available.

The staff believes that NRC approval of a complete set of design basis accidents would likely consider supplementing DOE-INL's proposed DBE-derived DBAs with deterministically postulated but physically plausible events.

So like on some of the previous issues discussed certain elements of the proposed approaches are somewhat overly risk-based and deterministic elements should be strengthened. And future Commission direction may be appropriate for some of these topics. No questions I'll go to the next slide.

So, overall to summarize while only looked at in the context of modular HTGRs the proposed approach is indeed a technology-neutral approach. It's comprehensive in that it considers full plant response to a wide spectrum of events. And the quantitative approach does enable the adequate assessment of safety margins.

The proposed approaches are generally consistent with past staff positions and Commission guidance. And I've got a few listed there, especially the advanced reactor policy statement, the MHTGR NUREG-1338, the technology-neutral framework and a few of the pertinent SECYs that have been brought up. And that is all I have.

MEMBER SKILLMAN: Question, please.

CHAIRMAN BLEY: Go ahead.

MEMBER SKILLMAN: Jonathan, a question. In each of these seven items DOE is requesting the NRC staff to comment on the direction that DOE is taking. Is that a fair statement?

MR. DEGANGE: I think that's a fair statement.

MEMBER SKILLMAN: Okay. In these seven or among the seven are there any differing professional opinions within the NRC staff and what conscious reflection has been given to groupthink for the staff's agreement to these seven items?

MR. DEGANGE: Well, I would --

MEMBER SKILLMAN: Number one, any DPOs.

DR. CARLSON: No, there were no DPOs. There was never any non-concurrence really. I think the reason is because we really haven't gone beyond very much past staff positions, ACRS comments, Commission-approved direction in these SECY papers, et cetera. We're advancing the ball just a little by providing a little more detail. We're really staying where the staff has already had staff established positions. So we were able to get to concurrence relatively easily because of that. It was a matter of how you say it more than what we're saying.

MEMBER SKILLMAN: Thank you.

CHAIRMAN BLEY: Don, I've got -- I didn't know where to ask this so I'll ask it now. You talked about -- and this is a concern to me because of this difference. Now we have white papers and we have an evaluation of white papers. The process is a lot different from having an application and having an SER.

When you have the SER, when you have all these RAIs if they're important issues they get reflected back in the revised application. Is there -- and you said that the RAIs were really more extensive at least in pages than the white papers themselves.

There's no process that pushes those back into the white papers. Is there anything -- well, I guess you've just got the big pile of them. I was just wondering would it make sense to somehow publish precis of the RAIs would be a companion to these two sets of documents.

I'm worried about us losing track of that useful information in the interim between now and when

-- should there ever be an application using this approach.

DR. CARLSON: Yes, I think you're pointing to the fact that as agreed with DOE-INL there was a decision not to update the white papers as we were going through this assessment process. And so the proposals that we're assessing are not exactly what's in the white papers.

CHAIRMAN BLEY: Right.

DR. CARLSON: In fact as one would hope through our assessment process they considered some of our feedback and clarified and I think modified to some degree their original proposals in the white papers. And so capturing that would best be done by some kind of future submittal. If not a revised set of white papers then a future submittal that reflects the staff feedback in future submittals.

Of course they don't have to take all of our feedback. It's just advisory. But they certainly should consider our feedback in developing future white papers, future submittals.

And I would point out that in our assessment reports, particularly the fuel qualification and mechanistic source terms, less so for the RIPB, we do have fairly clear linkage between each topic and the related RAIs. So anybody reading those topics would do well to go back and read each and every one of the RAIs and RAI responses associated with that.

CHAIRMAN BLEY: Okay, thank you. Before we take a short recess, break for the afternoon, I just wanted to turn back to the DOE and INL and ask if there are any short comments or issues of -- points of clarification you'd like to make at this point in time.

MR. KINSEY: Yes, I think we have two points of clarification not so much to cover in this meeting but that we'd like to see when we get the final output.

One is sort of a general item. A number of the bullets in the slides on event selection point to the fact that the staff feels more deterministic elements need to be rolled into the process. We're a little -- we're struggling a little bit to understand exactly what those are because the cutoff frequencies and the application process including the use of engineering judgment has generally been agreed to. So we just need a little clarification on what those additional elements are envisioned to be.

And then the second key item is the

question that I think you raised related to the need to add design basis accidents that are deterministically developed from a plausible evaluation sequence. We'd like to better understand how that process would work, where they would come from and what regulatory limits would apply to them once they're established.

CHAIRMAN BLEY: Okay, thank you. I think at this point we'll recess, take a short break. We'll be back here at 3 o'clock. Thank you all.

(Whereupon, the foregoing matter went off the record at 2:41 p.m. and went back on the record at 3:00 p.m.)

CHAIRMAN BLEY: The meeting is back in session and welcome back. I'll turn it back over to Don.

DR. CARLSON: Yes. So we're resuming with our feedback based on the feedback in the issue summary report. And so we've been through the first set of issues under licensing basis event selection. And now we're going into the issues under mechanistic source terms.

And Jim Shea has a very good background in source terms. He's been working in a more generic sense on source terms perhaps for IPWRs. He was a contributor in various phases of our work on evaluating the mechanistic source terms white paper. And so he's going to give his presentation.

CHAIRMAN BLEY: Excellent.

MR. SHEA: Good afternoon. My name is Jim Shea of the Division of Advanced Reactors and Rulemaking of the Office of New Reactors.

And just to clarify one point about the DPO. Our little slide there, mechanistic source term, I almost put a DPO in because Don wanted it to be called mechanistic source terms. And I told him that was wrong.

(Laughter.)

MR. SHEA: You can say that there's certainly not any groupthink when it came -- especially when it came to mechanistic source terms as even Stu Rubin who was part of a working group for a good part of 2 years. Those were very interesting meetings and I was kind of the ringleader of that circus. And a lot of disagreements and let me tell you. So it all kind of boils down to what we're going to show you here in the next 5 to 8 minutes. So next slide, Don.

First we'll start with definition. A mechanistic source term or MST is a best estimate analysis of fission product release from specific accident sequences including the necessary and assurance of plant and fuel performance as well as deterministic judgment to bound uncertainties. Now Stu, don't yell at me because that's probably not your definition that we came up with but it's just paraphrased from what we see in the slide. It shows that in SECY-03-0047 and 93-092.

The staff and the Commission approved the concept for mechanistic source term and in fact over -- since the early nineties the staff, the Commission including the ACRS have been receptive to the concept of mechanistic source term.

CHAIRMAN BLEY: And just for those of us who haven't followed this forever I assume what that compares to is just assuming some fraction is released without looking at any of the specific physical mechanisms or chemical mechanisms. Is that right?

MR. SHEA: No. We're talking about specifically a mechanistic source term. Deterministic source term you assume a certain fraction or the release from the core.

CHAIRMAN BLEY: You're saying the same thing. I was saying that was the comparison.

MR. SHEA: Right.

DR. CARLSON: Again, the definition of mechanistic source term in this case, it's event-specific.

MR. SHEA: Even.

DR. CARLSON: Yes.

MR. SHEA: And we'll get there. So slide 3. In general feedback the NRC staff believes that the proposed MST for the NGNP as outlined in its white paper are reasonable with some significant caveats.

And specifically what was requested in the July 6 letter to us was three main issues regarding mechanistic source term. And in addition to answering those three we're going to go into a little bit of what we said in the fuel qualification mechanistic source term assessment paper Rev 1 because we didn't want to be redundant in the summary paper. A lot of the information we already had covered. So we kind of briefly evaluated this feedback in the summary paper and then there's some more details. And we'll go over some of the details that we had in the -- the highlights I should say in the fuel qualification and MST assessment report.

Okay, issue 1 was the definition. The NGNP MST definition: event-specific radionuclides released from the reactor building to the environment.

Paraphrased.

The staff believes that the DOE-INL definition for NGNP mechanistic source term is reasonable and consistent with past Commission SRMs and staff SECY papers regarding the treatment for advanced reactors. Next.

These are pretty actually simple issues compared to the last stage. But issue 2, can the NGNP MST be event-specific? Mechanistically modeled and account for specific reactor design characteristics. The NRC staff believes that the described NGNP MST is reasonable and is again consistent with past SRMs, NRC and staff SECY precedents which is -- again was approved both -- in both SECYs the Commission approved the use of a mechanistic source term.

Issue 3 has DOE-INL identified the key NGNP fission product transport and associated uncertainties? The NRC staff believes the ongoing and plant testing and research activities for the NGNP FQ (fuel qualification) and MST (mechanistic source term) development are generally reasonable.

Here's where the rub is though. However, we do -- the NRC staff does expect more information on the issues of fuel qualification and accident testing and potential prototype testing as you might have seen through common themes that we've been going over as the staff.

MEMBER RAY: I've got a question. Just I can't put it off till the end without just becoming too distracted. Are we -- when you said mechanistic I thought we would include, for example, the issue that was brought up this morning briefly, that is, dust-related inventory blowdown. Is that included here?

MR. SHEA: Yes. And we're going to get to that very next slide.

MEMBER RAY: That's fine. It's just, it sounded like we were diverging off into something more narrow.

MR. SHEA: No. What we're trying to do is we give the real big picture and then we're going to go into some little bit of the detail, what the mechanistic source term is for the NGNP which is starting the next slide.

DR. KRESS: And when we did the mechanistic source terms for light water reactors we needed -- for the whole range of fission products we needed the volatilities, the LOCA pressures as a function of temperature. We needed affected diffusivities through various layers of the fuel and how to get out into the stream. Then we needed condensation rates. And then we needed how did they behave inside the containment in RCCS. That took a lot of labor. Do you see that you need this kind of data for the gas-cooled reactor stuff? That took forever to get all that data.

MR. SHEA: Yes. When you think of what you're talking about, referring to is really all the effort that went into NUREG-1465.

DR. KRESS: Basically, yes.

MR. SHEA: Which then culminated into the Reg Guide 1.183 which became the AST. And if you look at that effort and my colleagues here maybe could even chip in here if they want to, but if you look at that effort big picture they were taking some experience from TMI and they used that, and using MELCOR, et cetera, and trying to model that.

And then they tried to homogenize it over -- for all the different types of plant design. So there are some significant deficiencies in my mind as far as like plant-specific mechanistic source terms. Because in some cases one could argue in that effort that they were overly conservative in some aspects and maybe in another aspect not conservative enough.

And in fact not to go too detailed but you might remember in that effort we basically halted the core melt after 2 hours and didn't evaluate anything further. So the strength of a real mechanistic source term is to go beyond that type of like artificial deterministic cut it off because obviously we can't have a meltdown more than 2 hours, right? And look at each design and each plant individually and apply these fundamental physics.

And you know, obviously, as you know, as time has gone by MELCOR and other type plant analysis codes have gotten a lot better in characterizing this type of phenomenon.

DR. KRESS: Do you envision staff maybe working on the HTGR version of MELCOR?

DR. CARLSON: You may recall that the Office of Research has I believe reported to the subcommittee about 3 years ago on their effort to adapt MELCOR and other tools to provide an independent tool for the staff to use for a modular HTGR. And very few of those activities are continuing, just the work at Oregon State, the EPTF, the collaboration with the Japanese and that's about all.

MR. SHEA: Okay, so here's a little more detail of how NGNP would be treating its mechanistic source term. Now I have another picture courtesy of

DOE on the next slide that will just go through some of these attributes so we can just skip to that next slide.

If you look at this I think they referred to it as the pill model which I think I first saw that when DOE-INL first came and presented the HTGR concept to us. And I think it really depicts everything that's going on in a mechanistic source term model.

So we'll start with basically the fuel kernel goes through all the various multi-layered coating boundaries you can see depicted there. And then through the core graphite block.

Now, you've got to remember that the main safety case obviously is that most of the fission product radionuclides are all trapped within that in the fuel kernels -- or in the actual fuel particle. So 98.99 something. If you look in the white paper they have it. It's a very small amount of fission product actually is released.

And so then after the fission product makes it through the core and out to the helium pressure boundary which is depicted by the pill. And you can see the various mechanisms that would be evaluated in the model in a mechanistic source term.

And essentially so you'd have the model of how all of that fission product got out to the helium pressure boundary and then the helium pressure boundary and then the helium pressure boundary is a whole other problem in how it then is released.

And you notice some of the things we've already talked about today. There's plateout. We talked a little bit about the dust which is more of an issue associated with the pebble bed. It doesn't seem to be as much of an issue here.

MEMBER RAY: Why do you say that?

MR. SHEA: Well, DOE-INL have suggested that and I can give you some anecdotal evidence from some staff members who went to Fort St. Vrain after it was -- for decommissioning and couldn't find any radioactivity to clean. So that's just an anecdotal evidence that there really wasn't a lot that was produced even in Fort St. Vrain years of operation to decontaminate.

DR. CARLSON: I'll put a little caveat on that because as -- we did have a dust workshop with DOE-INL. And the predominant view there was it's probably not a big issue for prismatic block but there was a caveat on that.

In particular I provided a reference to a German paper, the chief chemist at the AVR, the pebble bed reactor Julich had a notion that the dust generated in the PBR was chemically produced. And we can describe that to you. But I translated that paper for the workshop.

And so a caveat, a note was made that that mechanism potentially could apply to a prismatic block reactor under certain conditions, notably if they have a high partial pressure of carbon monoxide and hydrogen in the helium.

MEMBER RAY: I've been to Fort St. Vrain too but I thought that maybe it was sensitive to flow velocities and the extent to which erosion might occur over a longer period of time than Fort St. Vrain was able to operate. Things like that.

In any event, any piece of information is useful. I just think it needs to be demonstrated that it's not an issue because the inherent presumption might be well, over time you will see erosion of the block prismatic forms. And if you're not, well why not. It's just a little --

DR. CARLSON: There are other mechanisms of dust. I mean oil can intrude into the primary vent and then that becomes dust. But where those were well discussed at the workshop. And so I think that provides a good basis for further review of these issues.

MEMBER RAY: It's a big deal. I mean because it's an immediate source term as opposed to something spread over days and weeks.

DR. CARLSON: Well, I think we have given due attention, we'll continue giving due attention to dust. So I'll turn it back over to Jim.

MR. SHEA: Yes, and well, just to follow up on that. I mean one of the things that DOE-INL are talking about as far as a conservative use of the mechanistic source term is to apply this buildup of whatever dust and plateout on the helium pressure boundary over the life of the plant and use that as their design basis accident dose source term. So that's a fairly conservative use of that whole concept.

MEMBER RAY: Yes and the flow circuits may have unique traps in them I suppose, things like that.

MR. SHEA: Yes.

MEMBER RAY: I have no idea what to expect.

MR. SHEA: Right. And I think it also falls on how would one -- if a plant would operate for 40 years how would you know that you're actually meeting these design goals. Well, I think we talked about it before but in the purification system and in actually monitoring the radionuclides in the helium pressure boundary you can get a good idea of what met those goals.

So okay. So the concept is as -- so you build up a source term essentially in the helium pressure boundary and from there depending on the event sequence, the specific event sequence that could be -- that would have maybe separate release parameters.

And then you notice that it gets released to the reactor building. There was a lot of discussion about the reactor building today. One of the key concerns is that then the release to the reactor building, it's the release from the reactor building to the environment that is the definition for their source term.

Now one of the treatments of the reactor building that would be an issue going forward is the credit for that decontamination factor that we were discussing earlier. And the staff has not ever credited reactor building for a decontamination factor. So that would be an issue going forward.

MEMBER CORRADINI: Can I ask you a question I guess? And maybe this has been -- maybe when it's appropriate for DOE because there's a couple of things the staff, the DOE staff and the contractors may want to comment on. But in this case the thing that's different about this reactor at least as I see it is there is no water. So it's a dry system.

So is the source term, the mechanistic source term that's being developed going to be developed mainly empirically by the DOE tests, not by any sort of calculational procedure?

MR. SHEA: Well, the tests are going to validate the empirical. They have empirical models to predict how the transport. And then there's the AGR testing that's ongoing.

MEMBER CORRADINI: Right. But in my simple mind there's enough parameters they're going to tune the models to what the tests show to make sure the models meet the test results.

DR. CARLSON: They have, in the AGR program there's a distinction between -- there's the early testing phase that focuses on providing data to develop the fission product transport models. And later in the phase the emphasis shifts with induced separation to validation of those models.

MEMBER CORRADINI: Right, and that's AGR-567 if I --

DR. CARLSON: Yes.

MEMBER CORRADINI: No. I'm remembering wrong. Can you at least tell me the numbers, Dr. Petti?

MR. PETTI: Eight.

MEMBER CORRADINI: Eight. Okay. That's the fuel qualification.

MR. PETTI: No, that's the validation.

MEMBER CORRADINI: Validation, excuse me. I'm sorry. Okay.

MR. SHEA: Okay, getting back. The key issue about the reactor building, the staff feels that a license applicant would need to provide adequate justification to credit the reactor building as a barrier for release of the source term. So that's an issue I think that's going to be ongoing for a number of the advanced reactors as far as taking credit for the reactor building.

CHAIRMAN BLEY: Have you done any work along those lines? What kind of -- any idea what they'll need to provide to justify?

MR. SHEA: I personally haven't done any work. Michelle, do you have any thoughts on where staff is heading? I promised I wouldn't pick on Michelle at this point.

MS. HART: I'm Michelle Hart from the Radiation Protection and Accident Consequences Branch in NRO.

And we haven't really done a lot of thinking about what they need to do. I mean they'll have to justify the paths that they're going to take, the kind of flow paths through the building and the deposition rates.

We've given credit for some holdup in some buildings where they have like a secondary containment where they have tech spec leak rates and they test it. I don't know what it's going to look like for a reactor building that may not have those kind of criteria. But they'll have to justify the models as far as some reasonable deposition rates.

MEMBER CORRADINI: Can I just ask a follow-on question? So we're talking for the DBA part?

MS. HART: For the DBA part, correct. To show compliance with the siting criteria.

MEMBER CORRADINI: Okay. With the siting criteria. And so I'm looking for an analog in the LWR. In the LWR there's a required -- if I remember correctly there's a percentage of the fuel that is failed within essentially a release based on the alternative or the older source term, right?

MS. HART: That's correct. And it's released to the containment and then a tech spec testable release rate from the containment to the environment.

MEMBER CORRADINI: But it's a tech spec testable release rate of the stuff inside containment with no decontamination of whatever. Whatever gets out of the core is assumed to be in the atmosphere.

MS. HART: We do give credit for decontamination within the containment. And also spray removal.

MEMBER CORRADINI: Okay.

MS. HART: So natural --

MEMBER CORRADINI: And all that is empirically testable. I'm trying to make -- I heard what Jim said. I'm trying to make an analog to what's done currently in the DBA for the LWRs. That's what I'm trying to make the analog clearly.

MS. HART: Right. The deposition rates are -- do have some empirical basis but as far as testing within a containment to determine the kind of deposition rates it's not really that. I mean there's an empirical model that actually Dr. Powers developed that we use for natural deposition in containments.

We've used that for large light water reactors and some of the current light water reactors as well.

MEMBER CORRADINI: And so there's some --

MS. HART: It's correlated --

MEMBER CORRADINI: -- there's some removal rate.

MS. HART: -- to power. Yes. And size of the containment. So it's a correlation there. It's not a first principles type model. But it's based on empirical data from small-scale tests.

MEMBER CORRADINI: Okay, fine. So just so I say it differently, from a natural analog standpoint given the fact that this design or these types of designs would rely on some sort of decontamination there would have to be a combination of testing and modeling and tuning of the model to essentially get some sort of credit for it.

MS. HART: And you may be able to build on those models that we already have or that we've already used in developing some of the models that we use. It's just we're used to giving credit for safety systems and for systems, leak-tight containments or secondary containments that are testable. What would that look like for this. I don't think we've quite gotten there yet.

MEMBER CORRADINI: Okay. And can I ask another question? A technical one. Which is to go back. I'm back to dust. The reason I guess I'm trying to struggle with it technically is that it seems to me at least for this sort of -- I'm kind of with Dr. Kress who had some comments from the January meeting where trying to bottle this up makes no sense in terms of a concept.

If you were going to have some sort of confinement or I'll call it controlled leakage environment or containment or confinement that's perfectly fine, but that means then the whole concept of what you have in there that you vent has to be fairly well known. Or you have to know the uncertainty band of it. So, because you're just going to release it. All right. And so that kind of goes back to what I think Harold was saying. Interesting in how the dust -- where it is and how much is there that might be the source of your vented inventory.

DR. CARLSON: That was the subject of that workshop that we had on dust.

MR. SHEA: Okay. So, next slide we'll go on to. I just want to highlight some of the highlights from the fuel qualification MST Rev 1 staff assessment paper.

And essentially staff again believes that the proposed approaches to the MST are reasonable with some caveats. And we really describe most of those in the fuel qualification MST paper.

But the highlights are the fuel qualification obviously in the AGR testing is ongoing. And still having plans in the future to do that.

Part of that AGR testing is the accident testing that will go on where they fail fuel and they actually test what the transport would be through some failed fuel particles. Also as part of that post-irradiation fuel testing as part of all that.

And also the idea of validating the codes and the methods and the mechanistic source term after a prototype reactors designs and use that as a means of validating the methods and codes and the transport in the mechanistic source term.

Another big issue obviously with this concept is to -- is the PRA, the quality of the PRA. It's going to be a little different obviously than what we've done in the past where we never reviewed it.

And we just, in our slide we talk about the peer review requirement in the ASME/ANS. And in addition the staff feels that the staff itself may have to review that PRA if it's going to be used for licensing purposes.

MEMBER CORRADINI: Can I ask on this one? It's the DOE's plan with the contractor to finish these fuels testing. So will staff come back and look at the results of that and remove some of these caveats? Is it the plan of the staff to --

DR. CARLSON: We don't have any plan. If there is an application for a modular HTGR then of course we would expect them to look to the AGR program for the technical basis for the mechanistic source term. Everything has been discussed here including dust.

MEMBER CORRADINI: Okay.

DR. CARLSON: And at that time then we would take a very close look at that.

MEMBER SKILLMAN: For a peer review of a PRA for an HTGR are the human resources available to conduct that peer review?

DR. CARLSON: That's an interesting question. I think we have seen that DOE-INL has brought together a group of HTGR experts, some of them, most of them from General Atomics. And they're all rather gray-haired. So the question is that hands-on expertise from actually having built and operated an HTGR, how much longer is that available. Interesting question.

That said, the program -- Oak Ridge work in the past and their recent work with INL is developing new expertise. So they can fill some of that gap.

MEMBER SKILLMAN: Okay, thank you.

MR. SHEA: Next slide. The last couple of things I want to point out on the fuel qualification MST paper are the staff feels that -- or I should say the staff believes that the siting DBA should include postulated bounding events which may include air ingress and water ingress.

In addition, the safety train study should be evaluated to inform selected LBE DBAs used in establishing the EPZ and emergency preparedness requirements.

Also, the DOE-INL research plan for air and water ingress that was recently submitted we feel represents a reasonable approach to addressing the issue of air and water ingress and its effect on the TRISO fuel particle.

And the last point is that the SECY-05-0006, the staff there recommended that for compliance source term should be based on a 95 percent confidence level. And that's I think been -- if you notice that's been a standard theme throughout all of our presentations.

MEMBER STETKAR: Jim, back to the first bullet. If I read that. That presumes that the PRA would not include those types of events.

MR. SHEA: I'm glad I had the opportunity to answer this question because I was sitting on the sidelines listening to this. And the thought is that the applicant, the licensee would have a complete PRA that would include not only all the events that we would consider for licensing which would be DBAs but also they would have done the safety train studies that we would be interested in seeing.

And I think of one specifically that the Commission requested for the last effort in SECY-93-092 where they requested basically a chimney effect. And during the last effort they evaluated that and even though the staff came back and concluded that that was not required for licensing purposes, it's just too incredible.

So, but that would be an example of where safety terrain on any of these advanced reactors, it would behoove the applicant to look at all those type of events. And the staff then -- that doesn't mean the staff when they -- when we get an application or in pre-application space wouldn't suggest that we may be missing something there to look in the safety train or maybe it's possible he has a DBA.

Because you think about it, you know, and your earlier question I was thinking if you look at how they do this license-based event selection it's really a deterministic effort. Because they have to go and actually pick the events. For example, start with station blackout and then run it through the models. Start with the LOOP event. Run it through the models and see how it goes and on and on. Start with the flood and see where that heads or start with -- or maybe even a safety train would be to say that we're looking at a seismic event that's, you know, 0.6 instead of -- some seismic event that would exceed licensing basis. And determine if there's a cliff edge.

MEMBER STETKAR: But in the way the PRA has been characterized at least in the white paper is that they're not discrete necessarily events the way you characterize them. They would look at the full range of seismic events, anywhere from zero g out to in principle 200 g if you could support that, or a couple of g for example peak ground acceleration which might have very low occurrence frequency but a very high consequence. They would look at all of those. So the scope of the PRA would indeed include all of those. They would include all of the possible fires in the plant regardless of the size and the location. You know, unless they were physically impossible.

MR. SHEA: I'll be getting more into the safety terrain --

MEMBER STETKAR: So that's where I'm getting in terms of -- you're characterizing it as safety terrain. I'm characterizing it as scope of the risk assessment.

And in that sense, you know, the risk assessment ought to include air ingress and water ingress events because one -- unless physics prohibits them from ever occurring under any condition the PRA ought to evaluate them with the frequency of occurrence, the consequences if they do occur both in terms of fuel protection and the ultimate consequences in terms of releases at the EAB.

So one ought not to necessarily focus on those events and say well you have to postulate those separately. They ought to be in there. If they're not in there the peer review both industry-related peer review and the staff's review ought to identify that as a deficiency in the PRA that should be evaluated. Once it -- if they refuse to evaluate it that's a different issue.

MEMBER RAY: Well, you aren't assuming this has to be a Part 52 application, are you? You may presume it would be. So it could be a Part 50 application. Any sensible applicant that's what they would do. But that's just my two cents' worth.

MEMBER CORRADINI: How does that --

MEMBER RAY: It has to do with what your expectations are in the application.

CHAIRMAN BLEY: But this is that option 2 I guess from the paper which says it's a risk-informed approach as well as the traditional approach.

MEMBER RAY: I saw that and I wondered at the time are they meaning to exclude Part 50 applicants. Because that's far and away the most likely application.

DR. CARLSON: The technical standards whether it's Part 52 or Part 50 are really in Part 50.

MEMBER RAY: So it's whatever is required for Part 50. Okay.

CHAIRMAN BLEY: Well, but they've modified it in this licensing framework to include a number of probabilistic approaches that aren't in the standard.

DR. CARLSON: So yes, the idea is to adapt the existing framework without perturbing it unnecessarily.

MEMBER RAY: Well, I'm just making the point, Dennis, that you're not going to have the same level and quality of detail in a Part 50 application that you would have to have in a Part 52 where you're seeking a certification.

DR. CARLSON: Well, by the time you grant the operating license it will be the same.

MEMBER RAY: Oh well, operating license is helmets. Start with the construction permit.

DR. CARLSON: Yes. You're done, okay. So I will --

CHAIRMAN BLEY: Is that all there is? That's it. Unless you want another question.

DR. CARLSON: Well, we're going to cover some of the same ground under this heading too.

CHAIRMAN BLEY: Okay, well let me ask a question because I didn't think about it earlier. In fact, if you flip back to your picture on page 30.

DR. CARLSON: And I said courtesy of DOE and DOE got that from Dave Hansen who created that many years ago.

(Laughter.)

MR. SHEA: I wasn't sure why it wasn't in their presentation this morning. I was looking for it.

CHAIRMAN BLEY: It's been there before. I was thinking about Dave Petti's discussion this morning about how they think a lot of the -- what they've measured is coming out of the matrix around the TRISO pellets that's accumulated during neutron exposure. And they're going to do the tests without having that matrix to see just what's coming out of the TRISO.

But when the source term is developed it's at least in principle to look at what's coming out of the TRISO particles, what would be coming out of the matrix, what would be coming out of -- if there's anything in the graphite or in the dust. All of those things contribute.

DR. CARLSON: Yes and when we're talking about -- if you have a large depressurization, you know, it's the circulating activity, the dust, the plateout, the washout, everything that happens. But then as Dave explained there's a delayed heat-up that takes a day or more to develop and during that heat-up you're getting -- you may get additional fuel particle failures but a lot of what you're getting is fuel that's already -- fission products that are already outside coated particles. Or that are in --

CHAIRMAN BLEY: That are --

DR. CARLSON: -- in kernels that don't have intact coatings and just the heating dries that out into the ultimately the reactor building.

CHAIRMAN BLEY: But even if you learn nothing has come out of the particles in the accident you still have this other stuff that's out in other places that you have to account for.

DR. CARLSON: So that's actually in --

CHAIRMAN BLEY: That accumulates --

DR. CARLSON: -- it's larger than the initial release from the circulating plateout and all that.

CHAIRMAN BLEY: Okay. That was all. Thank you.

DR. CARLSON: So, containment function performance. Okay, the proposed definition of functional containment, and I quote, "the collection of design selections that taken together ensure that first radionuclides are exchanged within the multiple barriers with emphasis on retention at their source in the fuel. And that ensure that NRC regulatory requirements and plant design goals for release of radionuclides are met at the exclusion area boundary." And so we considered that definition.

And they have also asked then for specific requests, specific feedback on three elements of this approach to NGNP functional containment. Feedback on the AGR fuel program activities for the fuel qualification.

On options for containment functional performance standards and a little more definition on how we would go about selecting events for plant siting and functional containment design decisions.

So issue 1 is AGR fuel program activities. And they have asked the staff to confirm that plans being implemented in the AGR fuel program are generally acceptable and provide reasonable assurance that TRISO fuel can retain fission products in the predictable manner. And they would like us to identify any additional needs for testing for other information.

So the overview of our feedback is that the scope of the AGR activities, mainly the fuel irradiation and post-irradiation testing and accident heat-up testing that are being planned and carried out are generally reasonable within the context of pre-prototype testing. And that leads ultimately to the point that there is additional data that would be needed for prototype.

The AGR program has been -- has achieved very encouraging results to date. I mean Dave Petti has said that for AGR-1 they irradiated 300,000 particles at high temperature, high burnup and high fluence and got zero failures. And he's talked about the results of heat-up testing to date. That likewise is indicating that they seem to be on target to meet or exceed the level of performance that the Germans showed 30 years -- or close to 30 years ago.

So yes, the early AGR irradiation safety testing results do show promise for demonstrating much of the desired TRISO fuel retention capability. But we would need additional data from the NGNP prototype to provide reasonable assurance of targeted fission product retention in the fuel.

And we have particular needs. One is for test data on fuel irradiated in an HTGR for effects of plutonium fission products, palladium and silver in particular on TRISO fuel particle coatings.

Also, related to that testing in the prototype to confirm NGNP core operating conditions and the ability to detect potential core hot spots and any of their affects on fission product retention and fuel performance.

As was said in the licensing strategy report to Congress that we would be licensing this under 10 C.F.R. 50.43(e)(2) such that the requirements for testing would not be able to be met prior to a fuel loading in the reactor and that the testing requirements would have to be plated by testing and surveillance in the prototype. This regulation allows the staff to impose additional requirements on the prototype in terms of siting, design features, operating limits during the testing period to protect the staff and the public during that period.

So, under issue 1 we have additional feedback on what constitutes a good fuel qualification and testing program. We think they have a good one but their description of it in our view could have been clearer. Under the heading of adequately deciding the fuel service conditions and performance requirements for both normal operations and accidents.

As we have seen in the TRISO fuel PIRT that the NRC did, Stu Rubin was a key factor in that 10 years ago. Dr. Powers and Dr. Petti were part of that PIRT panel. And one of the things that was noted in the PIRT was the importance of palladium in particular for its potential to interact with the coating layers but in particular corrode the silicon carbide layer.

Well, the two principal sources of plutonium and silver are from plutonium fission. The yields are 50 times more than from uranium fission or something like that.

So, we feel that because of the importance and Dr. Petti is developing new insights on the importance of palladium, because of the importance of palladium and potentially silver on performance of the TRISO fuel particle coatings we think that plutonium burnup should be something that is specified in your testing program.

We also have in the FQ MST report a section on potential effects of irradiation parameter path dependence. And Dr. Petti in his January presentation provided a scatter plot of what actual irradiation testing conditions they have for the various particles.

And you see that the emphasis is on the high end of everything, the higher end of the burnup range, the higher end of fluence, the higher end of operating temperatures. So the question becomes is it possible that if you irradiate fuel to high burnup in high fluence but at a moderate temperature that that somehow affects fuel particle coating performance in a way that's not addressed by the testing. And of course we have -- I mean Dr. Petti has developed some model, the PARFUM code that would provide some insight on that.

And so in our discussions with DOE-INL on this issue over the last year it was noted that we really don't have a good -- a design such that we could have a map of the parameters that an actual HTGR core would have to compare against that scatter plot.

Dr. Petti did show, however, in January an example of that and it did indicate that the -- much of the fuel would be at these less extreme irradiation conditions.

MEMBER CORRADINI: So, I'm sorry, maybe I missed it. You're saying that they looked at two extreme conditions. There could be some middle range conditions that caused -- I didn't appreciate what your --

DR. CARLSON: It's a question. It's a question that we don't have an answer to yet. And so it's just in terms of being thorough that the presumption is in their testing program that if you irradiate at these aggressive conditions you would have addressed all the other conditions as well.

MEMBER CORRADINI: So there's something not monotonically increasing about release rates. Somehow it would go through an inflection and I get more release in the middle?

DR. CARLSON: Just --

CHAIRMAN BLEY: The chemistry.

DR. CARLSON: -- are very temperature-dependent, right? The irradiation creep phenomena. And that's an important phenomenon in their PARFUM model.

MEMBER CORRADINI: But I have it right in my head. I just want to make sure I'm not missing anything. There's some set of conditions that I would actually get due to chemistry and time and temperature more release at a lower temperature. So I'd get some sort of effect like this.

DR. CARLSON: What we're talking about is preconditioning of the fuel for its performance during accidents. So these are the normal irradiation conditions.

MEMBER CORRADINI: Okay. So is there something about irradiation at less extreme conditions?

DR. CARLSON: Lower temperatures, for example. That conditions the fuel differently. I think that the popular wisdom, the conventional wisdom is that the more extreme conditions cover that but we need to see a technical basis for that.

MEMBER CORRADINI: So lead me through the -- it was on the slide before where you said 10 C.F.R. 50.43(e)(2) allows the NRC to impose additional. So your thinking is that potentially the prototype demo, whatever word you want to call it, would have to go through some sort of in-flight testing to find a better word to make sure you have confidence as you increase the allowable range of how it would operate.

DR. CARLSON: We see a potential need for that.

MEMBER CORRADINI: Okay, fine.

DR. CARLSON: It will be a continuing topic of review.

MEMBER CORRADINI: So let me reverse it. Let me pretend I'm the DOE and the contractor. So is there any set of tests that is finite and discrete that would relieve this other than essentially prototype power ascension testing?

DR. CARLSON: I think the overall message is there are a number of issues that get you into this regulation prototype testing, core operating conditions.

MEMBER CORRADINI: But I'm trying to get you -- you're very clever. I'm pushing you here and you're over there next time. So I'm trying to understand.

DR. CARLSON: I'm sorry, I'm sorry.

MEMBER CORRADINI: But what I'm kind of hearing is that save that there's very little, little compared to light water reactor, data you'd like to see essentially -- I'm reading it this way, that I'd like to almost see the demo built and watch it work its way through a series of in-service, in-flight testing to make sure you have confidence and your confidence builds.

DR. CARLSON: In the prototype you would want -- it's very difficult to do in core measurements in HTGR. In the prototype you could do that on a provisional basis.

Then I think the ultimate proof would be you would do PIE on fuel that's been irradiated in the prototype.

MEMBER CORRADINI: Okay.

DR. CARLSON: Over the full spectrum of parameter path-dependent conditions.

MEMBER CORRADINI: Okay. Fine.

DR. CARLSON: Okay. So again, as I was --

the potential for hot spots. I've been talking about hot spots a lot for years. People remember my references starting 12 years ago to the AVR net wire experiments and the hot spots that they revealed.

It's an issue that we consider applies to all designs. It's been -- received particular attention for pebble beds but we can talk about it. But I do believe that there are issues to be considered in this realm also for the prismatic block designs.

Okay, also we want to adequately identify fuel service conditions and performance requirements for accidents. The design information is needed to confirm the assumed lack of specific testing requirements for reactivity excursion events.

So that to us would be an issue to be considered once we have de-scaled design information to determine whether, for example, is a rod excursion indeed a plausible, credible event or is it really precluded by design features such that it's not credible.

As Jim alluded to we -- because some of our interactions on this DOE has followed through with a report, a research plan for moisture and air ingress. We think it's important to implement the activities outlined in that plan to provide the data needs on fission product transport for -- and fuel performance for bounding events that involve those phenomena.

So, what are the particular issues that we identified in relation to TRISO fuel performance? It's realtime versus accelerated testing. In accelerated testing do you get the time and temperature? No, you don't. It's always condensed in time.

And in an HTGR environment you have a harder thermal neutron spectrum and a higher epithermal neutron spectrum. The higher epithermal neutron spectrum gives you more plutonium breeding. The harder thermal spectrum for a given amount of plutonium gives you more fission of plutonium in relation to breeding because the HTGR spectrum peaks near the 0.3 EB resonance, thermal resonance fission -- fission resonance for plutonium-239 and -241.

So there was actually a report that DOE and INL developed on this to actually compare the plutonium burnup in the AGR-1 test. You see the AGR-1 tests are being performed in a water-cooled spectrum so they don't have that prototypical spectrum. And the differences were significant but not orders of magnitude.

Okay, another --

MEMBER CORRADINI: If I might, just so I -- so given the larger plutonium content we're back to the palladium and the silver in terms of how it affects the TRISO fuel pellet or fuel kernel?

DR. CARLSON: How it affects the coatings, in particular the silicon carbide coating layer. Palladium, there's been sporadic evidence over the years particularly in the Japanese literature of palladium attacking the silicon carbide layer and corroding it. Dr. Kania is an expert in this area. There's been some indications in the German program that they might have been starting to see some of those effects in the German testing. Certainly the INL program, the AGR program is looking at these and we think they should continue looking at these effects.

Another, you know, it's important to recognize. I was talking about hot spots earlier. That in an HTGR it's the helium. So for all gases viscosity increases with temperature. And these reactors have downward coolant flows. So the viscosity of the coolant and the buoyance effects tend to exacerbate the development of hot spots during normal operation. So that and the fact that you have -- there are various pathways for bypass flow between the blocks and the prismatic block. In a pebble bed and a prismatic block there's increasing flow through the reflectors.

As the reflectors -- over operating time the effects of radiation on the reflector blocks causes the blocks to shrink and so the bypass flows through the reflectors tend to increase with operating time. So those are all things that you consider under the issue heading that I call core operating hot spots, core operating anomalies.

Additionally you talked about in the prismatic block core you have a closed core coolant. You have individual coolant holes. And so you always have to add -- wonder if there's a potential for individual coolant holes to be obstructed in some way. And that gives you during quasi, ostensibly normal operation a hot spot. The question is can you detect that with whatever's available including monitoring circulating activity, plateout probes and whatnot.

Those would be things that you would have to actually demonstrate in a prototype and in a sense calibrate your plateout probes in your circulating activity.

Another issue was evaluation of irradiation temperature uncertainty. We heard that -- I think Dr. Petti gave a good discussion but we had an RAI that I wrote actually on they're having thermocouple failures during these irradiations and how is that affecting the uncertainties in the irradiation temperatures.

And so they provided a couple of reports that I noted there, especially the uncertainty quantification for AGR-1 and that is evolving for the different tests as Dr. Petti indicated in his presentation.

Bottom line, it's very important to understand how the irradiation temperature uncertainties are quantified and how they are affected if at all by increasing thermocouple failures.

The initial understanding from INL is that once you've calibrated and validated the way you calculate the fuel temperatures during irradiation the effect of thermocouple failures is not very significant. So that was to me counterintuitive. It will be an area for further review.

Another question was it was always kind of an article of faith that you could irradiate fuel in an HTGR or a test reactor, take it out and weeks or months or years later do these heat-up tests in an autoclave. And that would be representative of how the fuel performs in fission products transport in an actual reactor accident, either -- at power.

So you do have a hot spot. So at power does the fuel -- does the data you get on fuel performance and fission product transport in these delayed heat-up tests apply to at-power heating conditions, overheating conditions or the delayed heating that you get in the loss of core cooling event.

And the answer was always kind of qualitatively, well, it's the long-lived and stable fission products that dominate. And the short-lived fission products are not important.

And then there would be also the question well, maybe there are other things that change in that interim between irradiation and heat-up testing. You know, phase changes, something transports.

So I've been in the HTGR community in and out for decades and I never really saw a really scrutable analysis. So if you're saying that long-lived and stable isotopes dominate, that's something called the ORIGIN code can calculate for you, you can actually quantify that.

And so that's what -- in response to an RAI and some discussions on that RAI they did provide this report TIB-1543 by Jim Sturbens at Idaho. And it does actually provide quantified results. They do support the application of data from the lead fuel heat-up testing. We described that in the summary level in the FQ MST assessment report. And so we do have a scrutable case that does seem to support the case.

So issue number 2. The options for containment functional performance standards. Again this was the subject of some SECY documents in the SRMs. The SRM said to ask the staff to develop functional containment performance standards.

And the staff in an information SECY paper 05-0006 went back with the Commission to note the functional containment or the reactor building functions in addition to those for the functions we've been discussing under mechanistic source term. And those functions listed in SECY-05-0006 are protector significant, SSC, et cetera.

And DOE-INL, we all seem to agree that those are good functions. And we've added at the end this important one, limiting air ingress after helium depressurization accidents. So we think we have some degree of convergence in general terms of what functional containment performance standards should look like. That said we see this as an area where Commission direction may be appropriate.

CHAIRMAN BLEY: One thing I neglected to ask you as you began all of this. Since we don't have a follow-on response report from DOE are there areas in these exceptions you've been going through, all of you have been going through, where there's substantial disagreement or is most of this you're getting reasonably close and expect to see some of this in the following performance testing?

DR. CARLSON: In my eyes I think that we don't have any major divergence on this topic. I think we're -- we don't have full convergence but it's reasonable.

CHAIRMAN BLEY: Okay.

MEMBER CORRADINI: Can we ask the DOE do they have the same view of the apple?

DR. CARLSON: I think this brings it to --

CHAIRMAN BLEY: We'll give them a shot in a minute.

DR. CARLSON: I think this -- the big issue really, and Jim started talking about it during his talk, but they asked about -- they asked us for feedback on event selection for plant siting. So we agreed with them that the core melt accident assumed for light water reactor siting may not be adequate to monitor HTGRs and this is consistent with what the staff thought going back to MHTGR.

And so of course we would have to look at a detailed application, et cetera, to be sure of that but it seems that, myself included, I can't come up with an event that melts the fuel in an HTGR. And I've been trying. And a lot of people have.

We think it may not be applicable to modular HTGRs. That's a footnote. It's not necessarily a requirement. And so it's something that can be interpreted. And they have offered alternate wording. It's an event that would maximize -- a credible event that would maximize releases from the fuel to the reactor -- from the actual helium pressure boundary to the reactor building.

MEMBER CORRADINI: So this is -- can I just rewind, make sure I've got it right? So this is not the DBA.

DR. CARLSON: Well, whether you call it a DBA or not it's the event that you use for the siting source term.

MEMBER CORRADINI: In the world of 10

C.F.R. 100 and TIB-1484 it was called the MCA, the maximum credible accident.

DR. CARLSON: That's Jim Shea's term. Maximum hypothetical accident.

MEMBER CORRADINI: Is that what they called back in -- in `58 I thought it was called the MCA.

MR. SHEA: And they used that also, both those terms.

MEMBER CORRADINI: Okay. And it's the associated source term therein.

DR. CARLSON: So we can talk to the terminology but the ideas I think are clear.

MEMBER CORRADINI: So then if I might just make sure that I get it. Then this is not the discussion about how I select the DBAs from the licensing basis events. This is how I come up with some sort of stylized source term that the containment function has to withstand to meet either the regulation or their EPA PAG.

DR. CARLSON: The focus is on plant siting and Jim noted that if not the identical event the same kind of thinking there on these other areas.

MEMBER CORRADINI: Okay, all right. Fine. Got it. Thank you.

DR. CARLSON: So, the history is very important here. For the MHTGR and SECY-93-092 and NUREG-1338 is a very good document. I think DOE-INL has summarized that for you a couple of times now.

The staff employed some staff-selected bounding events with the idea that that would ultimately inform the selection of a siting source term that would determine functional containment that would be used in functional containment design decisions.

Well, the SRM on SECY-93-092 approved tat specific mechanistic source terms subject to adequate understanding of the fuel and fission product, and that's what the AGR program is all about, establishing that technical basis. In a sense in terms of policy the Commission says this is the way to go. Now it's a technical issue assessing how well we understand the fuel and fission product performance.

But in the SRM the Commission said your bounding events were good but we want to see more. And in particular they said, I'm going to read it, "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 graphite fire and a subsequent loss of integrity of the fuel particle coatings."

Now, people have reacted to that in various ways including wow, that is an event that's so improbable it's hard to put a number on it. I characterize that as a good example of the questioning attitude that we all need to bring to this. This is a technology that not everybody understands and the people that do understand it probably are going to understand it better after all this testing is done.

And so I guess, I've been talking about it in terms of the safety terrain. It's the questioning attitude. What would it take regardless of -- step back of what the PRA says and say what would it take. Let's postulate things.

And I can go back to we started doing that for the MHTGR actually before I joined the NRC. I picked up the reins from Pete Williams when he left NRC and joined DOE in 1991.

And I think at a presentation in this room, I think it was for the PIRT that we did for PBMR I presented some studies where we with Syd Ball and his code in the late eighties and early nineties, we looked at things. Well, what-ifs and what-ifs and we looked at what would happen if you had a rod withdrawal accident and no SCRAM and somebody turned on the helium circulators. Well, it turns out that according to the models at that time you could get fuel temperatures way up there.

Now, we're not saying that's credible, we're not saying that should be a licensing basis event but it's surveying the safety terrain. Out there somewhere, maybe it's not a cliff edge but there is an interesting terrain and now we need to understand what are the SSCs, what is the plant capability, what is the programmatic DID that keeps us far from there. And so we're viewing it in that light.

Let's look at things like this extreme air ingress event, the extreme moisture air ingress event. There have been published studies in the literature. Matt Richards, former General Atomics employee, Syd Ball, Oak Ridge National Lab, published some papers in the last 4 years or so that do studies of this type, extreme air ingress events.

I would put a caveat on that that all of the studies that I've seen today use the graphite oxidation properties for pristine unirradiated graphite. And it's important to note that contaminated graphite will tend to oxidize a bit more.

Pebble graphite isn't really graphite, so that tends to be more reactive than true reactor graphite. And irradiation damage to graphite tends to increase oxidation of the graphite.

MEMBER RAY: If you can have extreme air ingress, you've mentioned that a number of times. John mentioned earlier water. This being an embedded plant, a flood level sustained over some considerable period of time, water ingress, it seemed to be a possibility. If you're submerged for a week that's not -- it doesn't sound to me like a terribly incredible event for many sites.

DR. CARLSON: Well, I think the water ingress that we're talking about here is that for the MHTGR and for NGNP we're talking about a helium cooled reactor that has a steam generator on it.

MEMBER RAY: I know that. But I'm asking about a siting question since that was the essence of the discussion. And you talk about air ingress. Is there any reason why water ingress wouldn't be a problem given what you've said about there being a pathway for air ingress. What would prevent -- in a flood scenario what would prevent water ingress?

DR. CARLSON: Okay, well obviously as you heard there's the ingress from the steam generator.

Then to your question --

MEMBER RAY: No, I understand that perfectly well.

DR. CARLSON: To your question it's a perfectly good question. I mean one of the discussions that we had when Fukushima was happening is well, what would it do to the reactor vessel if you suddenly got a bunch of cold water from a flood.

MEMBER RAY: Yes. I mean we're talking about that we don't need a building that is LWR containment style as far as releases are concerned. I'd certainly subscribe to that. But of course you're giving up then, and you're embedding the thing below grade. You're giving up the protection that might be afforded by. I mean it's just a siting issue. It perhaps isn't relevant to --

DR. CARLSON: -- the safety implications of having water where you don't want it in the reactor.

MEMBER RAY: Yes, and lots of it.

DR. CARLSON: Yes, and lots of it. That's a very good question that certainly will be considered.

I would just point out that in talking about how you mitigate an air ingress some people say well, maybe preventing the air ingress by flooding with water.

MEMBER RAY: Well, that might be the answer you'd give. Yes, I understand that, right. But you can't only assume it's air or some limited amount of water from a steam generator.

DR. CARLSON: If there were an event that would give you moisture ingress then air ingress we would evaluate that. Or vice versa.

So we had -- this is not in their white paper. This is one of those things that we iterated on with DOE-INL over the past year or so. And they came up with the following approach which I think is reasonably how we would characterize it.

The applicant should submit for NRC consideration a risk-informed selection of siting events building on the types of bounding events considered by the staff in NUREG-1338 for mhtgr. And they've been presented to you by DOE-INL.

And to get to the SRM, the SECY-93-092 safety terrain studies if you want to call it that to assure that there are no cliff edge effects. Credible events. And let's not talk about probabilities, let's just look for the cliff edges. Credible events with high dose consequences. That's what a cliff edge effect is.

And to understand the ultimate safety capability we look at bounding event selection to be further informed by these exploratory studies of postulated extreme events. The rules of these exploratory studies should be that the events are physically plausible. You don't suspend the laws of physics. You do take credit for the inherent features of the modular HTGR design, the inherent behavior.

We think that is a path to informing the selection of the siting event. Does it give you the siting event? No, that depends on a lot of detail and a lot more discussion and possibly Commission direction.

MEMBER STETKAR: So are we hoping, by doing this though, the HTGR, the NGNP to a much, much, much more restricted standard than we do any considerable light water reactor that's either operating or being licensed today?

DR. CARLSON: That's an underlying thought that we've all --

MEMBER STETKAR: Here you go. I have to say this in every meeting. We don't require existing light water reactors to postulate a physically plausible event that I'll call an asteroid. Call it a meteorite. It's an external event. I've seen them. I know what their frequencies are. We don't require them to do that.

It's a physically plausible event that could result in a catastrophic set of consequences. We don't require them to do that because we say well, that's an acceptable level of risk on a frequency basis. So why -- we heard that frequency argument in the context of this particular design.

DR. CARLSON: Well, a light water reactor, certainly current generation light water reactor, there's no mystery about that. We know that the fuel will melt in those kinds of extreme events.

MEMBER STETKAR: I just use that as kind of an absurd example of extending that thought process.

DR. CARLSON: But the point here is we don't have that level of understanding. The modular HTGR design concept is basically an attempt to make it meltdown-proof. And so people naturally say hey, it's our job. We're the NRC. If there's a way to melt this or if there's a way to get a massive release, particularly one that doesn't take a lot of time to develop, it's our job to identify that.

And so does PRA the way we normally think of it with we don't go below certain frequencies, does that really give you that insight? To some degree maybe yes but I think to really understand the safety terrain you have to just say what if I didn't have whatever, a reactor building. What if somebody did turn on the helium circulator after you had a rod withdrawal event. What if somebody turned on the helium circulator, Stu Rubin will remember this, after a large break. That would be a way to get very rapid graphite corrosion, right?

And we had NRC delegations going to China to visit their HTR-10, their small pebble bed reactor. And in one of those visits Stu Rubin went in the control room, with the delegation went in the control room and Stu Rubin asked the host to ask the operators what they would do if they had a rapid depressurization and they saw the temperatures going up. And their response was we'd turn on the helium circulators. Wrong answer.

Actually that's what happened at Windscale. They saw the temperatures going up and what did they do? Well, that was an air-cooled production reactor. They blew more air on it.

By the way, if we were to update NUREG-1338 1989 version today I think we would want to correct a couple of things. There's a chapter in NUREG-1338 called graphite fires. And in that chapter they describe Windscale as a graphite fire. And I think you heard from Dr. Srinivasan among other people, probably Dr. Petti also that they actually looked inside the Windscale reactor about 6-7 years ago, have pictures, and most of the graphite is still there which reinforces the interpretation the experts had over the years that that was predominantly a metal fire and it did take some graphite with it. Likewise, the role of graphite oxidation in Chernobyl was not significant.

So I think by writing that section the way we did in NUREG-1338 we exacerbated this false perception that graphite and coal are the same thing. Dr. Srinivasan and others have shown you that it's very hard to get graphite to oxidize. That said, when you do have air ingress or oxygen ingress, hey, we're talking about cogeneration. One of the things that our PIRT panel for cogeneration identified was the potential for ground-hugging plumes, potentially oxygen.

So oxygen ingress can -- we're not going to say burn -- oxidize graphite. It's an exothermic reaction. And if you don't do something eventually it gets to the point perhaps if it's an extreme event where -- but it takes a long time to develop, where you could get increasing releases. So that's what SECY-93-092 SRM was asking us to do. We need to understand that better and I agree. The staff agrees.

MEMBER CORRADINI: So can I take John's point? Since he started so I won't give up on it. If I went back to the MCA and the originally TIB-1484 now where we are, now not in `58 but in 2013 I can run a MELCOR calculation and with the right knob adjusting I can create a source term inside containment that looks a whole lot like that first source term, right?

DR. CARLSON: Yes.

MEMBER CORRADINI: Okay. And then now by turning the knobs I have a sequence of events that gets me there which means I can actually compute a probability, a likelihood of getting it.

But your point is, I'm just trying to repeat your point, that we have no -- we have no MCA for this sort of design. You want to come up with one and you want to relieve yourself of the worry how I got there probabilistically, just how can I get there mechanistically. So I don't -- have I got it approximately right?

DR. CARLSON: Yes. Somewhere -- we don't

want to use probability cutoff necessarily. But we don't want to go to something that's beyond credible. So where is it? Well, we'll be informed by these cliff edge effect studies. And like you say, once you identify all the design features inherent.

MEMBER CORRADINI: So has DOE suggested something in this regard? Because I think I know now what you're getting at. And I see why the fuels testing program of INL is important to that because it'll tell you what really is physically possible given some sort of temperature thresholds. But has there been a suggestion as to what you'd use as essentially the equivalent of the MCA by the DOE?

DR. CARLSON: Their original proposal would be that it would be more risk-informed. What we've been talking about is well that's not what the SRM to SECY-93-092 says. And I think it was reasonable that they said what they said. We need to understand the cliff edges.

MEMBER CORRADINI: Okay.

DR. CARLSON: And so once we have that that will inform the selection of something that's credible. And where it is in PRA space may be very --

MEMBER CORRADINI: So let me take you a totally different place. When staff I think issued the equivalent of the SER for Clinch River that never got built the equivalent was a sodium fire for containment and an HCDA for the core. Is that what we're getting at? It's essentially -- because there was no -- we didn't suspend the laws of physics. On the other hand we didn't do a probability estimate of what would be essentially the Clinch River had to contend with for essentially siting. So that's what we're talking about. I hate -- pick another technology because this is going to come up again.

DR. CARLSON: Yes, I think we're talking about the same language here.

MEMBER CORRADINI: Okay.

DR. CARLSON: So, with that we can turn it over to Arlon Costa on emergency preparedness if there are no more questions.

CHAIRMAN BLEY: Okay.

MR. COSTA: My name is Arlon Costa. I'm a senior reactor project manager with NRO, the Division of Advanced Reactors and Rulemaking. I'll be addressing emergency preparedness as one of the major issues referred to in the DOE-INL draft summary document shared previously with ACRS.

I will follow a similar format as that of my colleagues where I'll discuss only NGNP pertinent background information, a summary of the staff's feedback on the issues and then I'll presents practice feedback on the issues.

The issues are related to DOE-INL's request for NRC to propose new policy or revised regulations and establish guidance on emergency preparedness requirements for emergency response plans and related issues such as emergency planning zones.

So I'll start with a background. We received this white paper that's in bullet 1 there that described -- that was submitted by DOE-INL in 2010. And it contains important information obviously addressing emergency preparedness, specifically information on the modular HTGR core.

It provides information as we've seen extensively today on the TRISO barriers that are associated with radiological releases and many other important information. All that is in support of a smaller size emergency planning zone when compared to current EPZ's required for light water reactors.

The NRC did not provide a formal feedback to DOE-INL on this white paper submittal but emergency preparedness framework issues were addressed in various public meetings. And the main reason for the postponement of the formal review was that key EP staff resources were prioritized for other projects and later on to Fukushima.

But 1 year later in October 2011 the staff issued the policy information SECY paper mentioned there, SECY-11-0152 titled "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors."

During the development of this SECY paper the staff considered information as outlined in the above described DOE-INL white paper. And also before issuing the SECY the staff had discussions with various governmental and private sector stakeholders in discussions related to alternative EP frameworks.

So regarding the SECY it focused on an EP framework that describes a general approach to scalable emergency planning zones. And we'll talk about that a little bit later.

So but let me give you now a short summary because we don't have time on the described EPZ categories that the SECY states. It states that EPZs could be based on a radio distance from the source and that the use of conservative calculations of the postulated accident dose can be defined by the actionable lower limit of the EPA PAGs which is 1 rem.

The SECY also contains discussion from

other critical interrelated issues impacting EPA requirements such as we discussed today a little bit on source term and also it takes about modularity and process heat applications.

So just quickly as a past historical perspective I would like to mention that emergency preparedness and planning included small reactors that operated in the U.S. And we are all aware that they had 5-mile EPZ. And they were obviously Fort St. Vrain which was an HTGR but the other two were BWRs, Big Rock Point and La Crosse.

But what is important to state here is that the fact that the regulations in 10 C.F.R. 50.47(c)(2) allows for an emergency planning zone size for gas-cooled reactors to be considered on a case-by-case basis.

I'll discuss further in my presentation but just a quick summary. And maybe I can use the favorite word of the day before addressing the EP issues which were presented to the NRC, DOE, INL by saying that DOE-INL's proposed approaches are reasonable. So that's the favorite word of the day. You've heard this probably 90 times today.

And also that the proposed approaches are responsive to the Commission's policy statement on advanced reactor. Now, that doesn't mean that there aren't policy implications. So plain and simple we're basically saying that it's important to emphasize that modular HTGR issues could have policy implications that would require future direction from the Commission as appropriate.

As far as alternative to EP requirements as a feedback the staff is open to considering alternatives. EP requirements and framework for advanced reactors and in fact the SECY that we wrote was for small modular reactor facilities. And we kind of included NGNP in some of the writing in there.

But here's a key feedback on the DOE-INL's request. And their request was that NRC establishes new policies, guidance, or revised regulation related to NGNP EP. Here's the feedback.

The staff does not plan to propose additional new EP policies or to revise the existing guidance for addressing NGNP EP requirements at this time. And I'll discuss this a little bit further.

Starting with what we call issue 1 DOE-INL requests that NRC proposes a new policy or revised regulation on how EPZ size can be scalable. And this is also seen today and throughout this presentation that their goal is to justify a 400 meter exclusion area boundary for the EPZ.

The feedback that the staff is providing is consistent with the policy information SECY paper that I just described. Let me additional a little bit more on the SECY paper that we wrote. As a broad summary the staff focused it on small modular reactors, namely integral pressurized water reactors. But obviously appropriate information was provided there in the discussions that were relevant to NGNP.

And reiterating once again the policy information SECY describes a dose distance scalable approach to determining emergency planning zones. This paper also discusses how emergency preparedness requirements can be simplified by applying a graded approach to addressing guidance used in demonstrating compliance with existing regulatory requirements.

So, the staff does not intend to offer anything further. And let me repeat the words in the summary report that we sent. It says that the staff does not plan to propose additional new EP policy or to revise guidance for specific changes to EP requirements at this time.

But as far as future proposals is concerned in the other bullets the staff is open to considering further proposal from industry or established pre-applicants. So topics for inclusion in the NGNP proposal are described in the last bullets. Actually the dash is there and which is related to PRA approach. I'm not going to read it to you. And the other one is the risk-informed criteria associated with the EPA PAG values which is talking about an acceptably low value.

So this way of looking into these future proposals is consistent with a purported reduced risk associated with modular reactor HTGR design.

CHAIRMAN BLEY: So this is really not saying anything new. It's just saying if you want to submit something that's new we'll look at it. Right?

MR. COSTA: That's a good way to say it.

CHAIRMAN BLEY: Okay.

MR. COSTA: Okay. On issue 2 DOE-INL's request that NRC establishes -- and these are key words -- establish specific guidance on EPZ graded approaches to applying to EPZ requirements in relation to the EPA PAGs. So the staff assessment is that NRC expects proposal from an NGNP applicant just like you just finished saying. But the proposed approaches should include -- and now we're talking about the considerations to be supported by the details of the design, the site and co-location use of facility.

We'll talk about co-location a little bit later.

And also considerations on how potential emergency preparedness basis events may be influenced by co-location and external events impacting the site.

Another consideration, and we mentioned prototype earlier, is that an EPZ graded approach may be different from an NGNP prototype plant as compared to subsequent standard plants. So the staff also makes it clear that NGNP EP approach addressing the EPA PAGs must be developed by the site applicant. That is because emergency preparedness is an operating license and a combined license issue holder.

And I also want to make it clear that obviously future Commission direction may be appropriate to address these NGNP frameworks. As you said, as they come to us.

CHAIRMAN BLEY: A question not related to NGNP but related to the other work you're doing which you put on hold unless you get an application for an SMR. When you do that -- I guess that would fall the same way. When you look at establishing review criteria for an SMR using the technology-neutral sort of approach would this still apply for EP that it's only if they include something on EP that you would look at, or might there be a review of those issues more generically to support that work? Is that something you can talk about now?

MR. COSTA: Oh no, in general EP is a defense-in-depth program. You know, so as far as the SECY paper we mentioned that in the SECY paper about technology-neutral. And I think Don also mentioned something about association with Nuscale.

The real point is that it's open. I mean the EP emergency preparedness is here to stay and we're basically providing an opportunity for this scalable approach that they can come and present to us to address emergency planning zones. But keeping in mind that emergency preparedness requirements is an overall -- has an overall umbrella into all the programs of preparedness, response and everything else that comes with the defense-in-depth program.

But this, the last issue is the issue that has to do with co-location and DOE I know requested that NRC propose guidance on how this works out for multi-module plants.

The staff responded to this request to propose guidance by noting the co-location similarity to existing water reactor plants. And noteworthy, I'd like to mention, Waterford which is located near an industrial park in Killona, Louisiana.

So in this case, and I'm separating the two cases here. My thoughts will come through right here. That emergency preparedness co-locations for current light water reactors are largely applicable for NGNP. And in case that the plant is solely designed to produce electricity and that it's co-located by nearby facilities. So that's one way of looking at it. So in this case regulatory guidance are already incorporating -- incorporated into existing emergency preparedness plans.

Now the other case, and you can see probably in your minds you've seen the NGNP cartoon of the steam or the power going in different directions. So that's what I'm calling here the coupled mode for cogeneration. This mode implies that co-located NGNP modules are utilizing nuclear heat byproducts such as steam to be subsequently used by industrial facilities. In this mode it can potentially produce electricity as well.

This NGNP cogeneration coupled mode carries a different regulatory nexus for emergency preparedness. And so therefore emergency preparedness must consider challenges and issues arising from the modular HTGR being coupled to the industrial facility.

Let me just mention some challenges that are related to this. For example, shared industrial facility SSCS, standoff considerations, potential explosions and missiles or fires, external events, the effect of chemical gases and radioactive hazards from industrial facilities.

And here are other interesting challenges that are related to this. Response coordination with the co-located industrial facility with the state, federal and county agency. And here's another one related to the resolution of jurisdiction issues associated with radioactive material monitoring and plant security. So you can see there is a gamut of challenges that could be there.

MEMBER STETKAR: Arlon, how did the Agency address all of those issues back in the mid-eighties when the Midland Plant was being licensed? Midland, Michigan.

MEMBER CORRADINI: And was operating.

MEMBER STETKAR: No, it never operated. But it was being evaluated to be licensed. It was a nuclear plant that --

MEMBER CORRADINI: -- steam to --

MEMBER STETKAR: Supplied steam to Dow Chemical, exactly your second sub-bullet under there.

MEMBER CORRADINI: In fact --

MEMBER STETKAR: It was -- that's what it was designed to do, Unit 1 anyway. Co-located is -- Dow was across the river but Dow was the biggest chemical plant I've ever seen in my life. I'm sure there are bigger ones elsewhere in the world.

But my own point is that in principle the Agency has addressed or should have thought about that issue anyway 30 years ago. And I was wondering whether you looked at any of that history in relationship to this current discussion. Or whether those concerns were even addressed at that time.

MR. COSTA: That's a good point and I must admit that I'm not familiar totally with -- I know what you're talking about, I have a recollection of it, but I did not --

MEMBER STETKAR: What happened was it basically was a B&We plant and they were delayed so long that Dow eventually figured a different way to get heat.

DR. CARLSON: If I remember correctly in our PIRT report for -- the one that was developed. There were several PIRTs that were done about 5 or 6 years ago DOE and NRC together, and I facilitated a panel on co-generation and NGNP. I believe there's discussion of that in there as kind of background information.

MEMBER STETKAR: I'm just curious because it would seem that there would at least be some regulatory discussion anyway if not a firm precedent.

CHAIRMAN BLEY: Can you remember the name of the river?

MEMBER STETKAR: Tittabawassee.

CHAIRMAN BLEY: Yes, me too.

MR. COSTA: So, but the two different views as you can see, one of them is pretty clear and the other ones we need to look into it.

You know, in fact in summary for NGNP co-location we need to expect staff considerations of new regulations. Especially as we look into lessons that we could have learned from the past, hazards assessments, accident evaluations, and security issues, all these things may come up.

In fact, we have an example in our SECY-11-0112. And there's a section there that we write on industrial facilities using nuclear-generated process heat.

And it states that any effects of the industrial facility on the reactor will be addressed as part of the NRC staff's review as part of the offsite hazards analysis. So we're not totally unaware of what's going on.

And then in fact on SECY-11-0152 we do mention NGNP and a statement related to that. It says that the staff does not have sufficient information at this time to determine who to propose emergency preparedness frameworks. They might need to be adjusted. So it's one of those things that we need to wait for that applicant.

MEMBER RAY: Well, but the applicant is waiting for some better idea of what the NRC is going to require. And so you've got a chicken and an egg situation here for which there seems to be no solution.

Is the problem that there isn't an applicant, or is the problem that you don't have specifics sufficient to answer the questions that DOE is trying to answer?

MR. COSTA: It's both. It has to do with the design, the specificity on the design and obviously it's chicken and an egg. They need to --

MEMBER RAY: Well, we have criteria otherwise that don't depend on having an applicant or a specific site. But they depend on --

MR. COSTA: Right.

MEMBER RAY: They state what the

requirements are to provide adequate assurance.

MR. COSTA: Right. Well, there is an element of timing here also as I stated before. Since this is an issue for operating license or a combined license it's something we can look into the future.

MEMBER RAY: Why do you say that, by the way?

MR. COSTA: Because that's the nature of emergency preparedness.

MEMBER RAY: Okay. So I mean this wouldn't be, you know, basically what you're saying is that you can't address these criteria just with a Part 50 application that has a site but doesn't satisfy all the information requirements for an operating license.

MR. COSTA: Well, an applicant has an opportunity to provide emergency preparedness information even during an ESP application.

MEMBER RAY: That's right.

MR. COSTA: So, but the staff would need information in order to assess the design and the specificity on the designs in order to understand things as we discussed. In fact the whole meeting today came to this point of emergency preparedness, everything that was set up to now enforces the issue of emergency preparedness. And we need to understand all that in order to come up with emergencies. Especially if you want to have a scalable emergency planning zone and then you want to be meeting the EPA PAGs. So it's pretty challenging.

MS. BRADFORD: This is Anna Bradford. I would just make one point about the chicken and the egg situation. I don't think it's quite a chicken and egg just because we do have EP regulations and guidance on the books. If they wanted to just meet those we wouldn't even need to have the discussion. They want to do something different. So if you want to do something different tell us what you want to do.

MEMBER RAY: Well, precisely.

MS. BRADFORD: Right, and we'll evaluate that in the context of our regulatory framework.

MEMBER RAY: Yes, well they have tried to do that I guess and it falls short of what you need is the best I can figure out. But the point is that, yes, there are requirements so why can't we have requirements that apply to something other than what the existing frameworks were developed for.

And you know I realize you have to have a certain amount of information in order to do that. But on the other hand it almost seems as if we're not prepared to do anything until somebody appears in the form of an applicant. But that's not going to happen until they have some better assurance of what the answer's going to be. And that's why I called it a chicken and an egg.

MS. BRADFORD: I think I would argue it the other way and say I'm not sure we would spend our resources to develop a proposed alternative, and that alternative might not be what the applicant wants to do either.

MEMBER RAY: That's, you know, that's fair. I mean it does take time and effort to do this. That's why we're all sitting here instead of my being at home or en route here as I would otherwise be. But the upshot turns out to be that we're basically not achieving anything productive. And I guess that's the conclusion you have to draw from it.

MR. COSTA: well, there is a path. One of the things that we coded was the case-by-case basis. So that was always there available.

And then as part of this progression we credited the white paper that they wrote and then after we considered that in public meetings we also provided to the applicant an approach that can be scalable specifically for EPZ. And then what we're basically said is reinforcing what Anna's saying. I mean we do have regulations that we can license in HTGR but there is possibly a better way. So just come and tell us and we'll be ready to listen.

CHAIRMAN BLEY: Well, I think, you know, when this whole business came out of the law saying DOE and NRC work together on this. And in most of the areas you've taken a look at their proposed framework and said yes, we kind of agree with this with a few exceptions for you to fill in.

In this area you kind of said well, submit an application and then we'll look at it. So it seems like you treated it a little bit differently than you did all the other issues on the table. Am I missing the boat on that?

DR. CARLSON: We didn't say submit an application because the advanced reactor policy statement says you engage before they submit an application. Pre-application review. So when we're in a pre-application review whether it's -- we expect a submittal and some plant item at a time. That's the time to engage. But that does require some level of design information, some specific proposals.

MEMBER RAY: Well, that's what I guess I'm saying is that you don't have enough that you can respond to even hypothetically. And it may be that your hypothesis would be -- not suit the applicant. But anyway, as I say, the aim here, the SECY requirements are probably if not the most important, one of the most as you know. For this thing to get any prospect of going forward.

And I still, I'll stick with my chicken and egg metaphor.

CHAIRMAN BLEY: But at this point why don't we go ahead.

MR. COSTA: Okay. I turn it over to you, Don.

DR. CARLSON: So that's all we had. Of course if you have other questions we have backup slides but I think we've gone through everything that we planned to present.

CHAIRMAN BLEY: Thank you. At this point I'll ask if there's anyone, a member of the public or otherwise here in the room who would like to make a comment. We'll listen to you at this point. And we're checking to see if there's anybody left on the line and if anyone else has comments.

DR. CARLSON: We didn't ask Mike Kania for his expert insights but I appreciate his being available and I hope he found it a good meeting.

CHAIRMAN BLEY: Okay. Mike, are you there on the line? Can you hear us and can we hear you?

MR. KANIA: Yes, I am on the line. I appreciate it. Thank you.

DR. CARLSON: If you feel that we failed to say something essential you can say it now.

MR. KANIA: The only comment I wanted to make is that you were asked earlier on that you know, all these RAIs were generated such and there was interaction between NRC and INL, DOE and INL, on these. But I think there was an excellent interaction.

And where you see the changes kind of effect is in the program plan that the DOE is looking at. I mean they're doing things nowadays that just wasn't possible 20-30 years ago. They're systematically doing the math balances, they're using techniques that were never applied before. The amount of mileage we're getting out of the R&D nowadays is just, it's just orders of magnitude more than what we got previously.

CHAIRMAN BLEY: Okay, thanks Mike.

Anybody else on the line care to make any comments? Then at this point I'll ask members of the subcommittee to offer up any of their comments. I'll start with Mike Corradini. Oh, I'm sorry.

MEMBER CORRADINI: So I cede some of my time to DOE and their contractors about some of the questions we asked.

CHAIRMAN BLEY: You don't need to cede time.

MEMBER CORRADINI: I know, I'm just joking.

CHAIRMAN BLEY: I did say I wanted to give you folks a chance to make any closing comments you'd like if you want to make them.

MR. KINSEY: I don't have anything. I think Dr. Petti may have a couple of.

MR. PETTI: Just I did not get into details but some of the issues that you have raised about path dependence and temperatures, the irradiation. Unfortunately you won't be able to attend our R&D meeting but you'll see that we have incorporated those in AGR-5/6/7 that's going to have a much broader irradiation envelope than say I think 10 years ago when we started. Stay tuned. It's very active and interactive.

DR. CARLSON: So I don't know how much that would reduce the scope of prototype testing but that would be good information.

CHAIRMAN BLEY: Okay, thank you.

MEMBER CORRADINI: So I have two things. The first one, first of all I think DOE and their contractors have done a nice job of reminding us what we heard in January. And staff, I appreciate their responses, very specific responses to some of what the DOE had asked.

I guess there's two things that came out of this that I hadn't put it in perspective but I'm looking at the chairman for the intent of something to put in the letter. So two things.

So first, provocative which is it seems to me with all due respect staff is back in 1958. You're basically saying I've got a siting study, I've got an MCA for the siting study and I don't know how I got the numbers but show me an MCA. And now if I now move forward 55 years to where we are we don't want to use any sort of risk insights as to the probability of these events to rank-order or think through this. To me that's interesting, surprising, not appropriate. Okay?

And the reason I'm saying that is if I put the same sort of argument for light water reactors and I went back to the TIB-1484 and I said what did I ask licensees at the beginning of their light water reactor generation to do. And then I now have the ability to calculate and see what's the frequency of that accident sequence that gave me that it would be at some number.

And I'd be willing to bet the frequency to get an engineering judgment extreme event for the HTGR, whatever it's called, the gas reactor is going to be a frequency not at that same frequency level but much lower. And to say that we're not going to at least acknowledge the fact that it's less frequent, of lower probability or at least have something as that as part of the mix surprises the heck out of me.

So I thinks somehow in the letter we have got to express the need to say we ought to have pretty much of a fair comparison to what we can analyze relative to both the consequence as well as the probability. Because I do think staff is right, there is not as mature as the light water reactors today, but I don't think we have to go back to an approach where we kind of just use engineering judgment without calculational expertise as to what the frequency of some of these events are that lead us to these. That's one. Two.

The second thing I think the staff said that I still don't completely appreciate but it really is something that I think DOE has got to worry about is that if the fuels testing program is very good or extremely good but still is lacking in some area I hear staff saying, I could be misinterpreting, that you're going to have to do some sort of ascension testing with the first one of these which makes some plausible sense.

But it seems to me that's got to be discussed with DOE now or otherwise they're going to leave here when we close all of this because of lack of funding going forward and they're going to expect X and staff is going to expect Y and there's going to be a divide there. And anybody that comes back to this that actually wants to build one of these as a prototype are going to have to do a whole series of ascension testing.

And I just sense a bigger gap there than maybe I first expected. And it may be just as I wasn't paying close enough attention.

DR. CARLSON: I would like to say something if it's okay that I probably should have said early on. Mike Mayfield has been very clear about this whenever he has a chance to say something.

MEMBER CORRADINI: I'm sure he has. I know him.

DR. CARLSON: And the vision is let's

figure out how to get the prototype to exist. So as far as we're talking about the first of the kind prototype, whatever you want to call it, the lack of maturity, the PRA technology, it's something we have to keep in mind.

And once you get the prototype up and running that -- and there are no huge surprises. And you learn something by doing actual measurements and testing. Well, maybe we can back off on some of these conservatisms.

Like I said before, if we were talking applying similar ideas to a light water reactor then maybe something like an option 3 would be appropriate. So again, Mike Mayfield says over and over again let's figure out how to license the prototype. And then the licensing basis for the standard plant may be adjusted based on those insights.

MEMBER RAY: Yes. I'd make the prototype of Part 50 application. But who's going to pay for it is still an open question.

MR. SHEA: I just want to address another issue brought up about the source term in 1958. I actually think we're probably closer to 1972. I've got my bell bottoms on. But in reality in 1958 we assumed the core melted without a lot of knowledge on how and it was a safety case that designed a containment so that you would protect the public.

So if you look at what they're proposing here and that the staff is accepting is a mechanistic source term that says there's no way that we can melt the core. That's a tremendous fast-forward. That's definitely an advance of thinking over the years.

So, and combine that with the fact that we're not saying the PRA can't be used. In fact no, we're saying that in fact the PRA is a strength because what it does in reality is it does not buy us the defense-in-depth in terms of all prevention. It puts some into mitigation.

In the past, in 1958 we put all the eggs into prevention. And we got a lot of accidents because of simple things like the loss of offsite power that could have been -- if it was evaluated under PRA and saw the strengths in both the prevention and mitigation those things might have been prevented. So no, I think it's definitely a modern look at how to evaluate reactors.

And I'd also comment that I think what we have outlined in our assessment given that there's a lot of like, you know, of course words where there's no policies, et cetera. However, if you look at it we've provided a clear path in my mind on how to license one of these reactors or any advanced reactors.

CHAIRMAN BLEY: Okay, thank you. Mr. Skillman.

MEMBER SKILLMAN: The concerns that I had were pretty much addressed when we went through the discussion about the strength of the probabilistic and deterministic methods being used. Your example of the rod ejection and actually needing to get a design up and operating in order to have greater understanding of how the machine is going to behave.

I think that there are some real challenges that lie ahead and that is seeing how the TRISO fuel operates in a real situation. And I think that the complexities of operating multiple reactors at a single site brings with it operational issues concerning staffing and attention to detail, how multiple units might behave individually and together.

I believe the emergency planning problems -- I shouldn't say problems. The emergency planning challenges will be new for this new type of reactor. We continue to discover new issues even in the current fleet relative to emergency preparedness and emergency planning and interaction with the state and local and federal authorities.

This will be a brand new learning experience should one of these ever be built or should a group of these ever be built. And so I think there needs to be extreme attention to detail because in a way this is introducing a new reactor thank you to civilian users.

I believe that even though there have been prototypes, there have been other gas reactors, this will be new. And so there needs to be a sense of caution that accompanies progress for this reactor type. Thank you.

CHAIRMAN BLEY: Thank you. Mr. Stetkar.

MEMBER STETKAR: Yes. I'd echo Mike's concerns about however you want to characterize deterministic versus probabilistic approaches to maximum credible accidents. He's much more eloquent about these things than I am.

A couple of other things, and I mentioned them earlier, just to kind of reiterate. And that is a bit of a concern about what I still believe is an inconsistent approach to addressing the effects of uncertainty and the consequences among the three or four depending on whether you consider DBAs a different category of licensing basis events.

And something I mentioned pretty quickly because it gets into a lot of details but I think there's some need for improved clarity on the notion of what event sequences are and how that notion of event sequences is and will be used to identify and categorize licensing basis events. Because there seems to be I don't want to call it a disconnect.

There seems to be a gap there. Because I can read words that sound like they say the right things but I see examples that might not implement those words. So with that that's basically what I have.

CHAIRMAN BLEY: Okay, Harold.

MEMBER RAY: Well, I certainly buy the idea that the best path forward is a prototype that doesn't have to solve all of the problems as if they were being solved for all time and that isn't dependent upon the usual commercial considerations in order to move ahead. That may be the only way that we can deal with some of these issues.

I think there's a general feeling that there are inherent safety advantages that should be recognized and particularly with respect to the ability to site a plant like this. But there are also obviously unanswered questions as well.

And at this point in time to be committed as the regulator to recognize those advantages without knowing more of the details is probably a bridge too far. So I hope DOE feels like it's been a worthwhile exercise to this point. But it certainly falls short of what some had hoped we'd be able to accomplish.

Having said that I can't suggest what we might put in the letter. I'll have to think about.

CHAIRMAN BLEY: I'd appreciate any thoughts. Tom, I know you're going to send us a report but if you want to summarize those vast number of comments I'd appreciate it now.

DR. KRESS: I would like to say that I'm glad to see this exercise going on because I consider the approach an improvement in the way we regulate licensed reactors. And that it's superior to what we've been doing.

And I think it's more what's -- I see it is more closely associated with option 3 in terms of probabilistic versus deterministic. That doesn't bother me because as best I recall the ACRS letter we recommended option 3.

But anyway, it really doesn't bother me. I don't know how to separate out deterministic versus probabilistic too well anyway.

I do agree with a number of comments, Corradini's and Gordon, but especially Gordon. I worry about one control room and one operating people for modular reactors. I don't know if that's a problem or not but it's something we're not used to and it could give operational problems. I don't know what they are and I don't know if we'll ever find out. But I kind of agree.

I particularly agree with John Stetkar's issues with the questions of inconsistencies in using the uncertainties. And also the clarifying what actually is a sequence is. When I first read all this I thought the sequences were every sequence I'd get out of a PRA. That's a lot of sequences and I don't know if they bundled them or not. But for gas-cooled reactors you don't really have that many sequences. You might be able to just look at all of them.

But I had some questions about the top-level regulatory criteria. Because you do have to specify up front how many modules you're going to have before you can see whether you're approaching the top-level criteria.

And I think the tendency would be maybe to choose those number of modules that keeps you below the criteria but gets you up close enough to it. And

I don't see any criteria on how close we ought to allow in terms of uncertainties.

There was some on selecting SSCs. And I also worry about that because I think having a stair step top-level regulatory criteria can affect the SSCs you come up with depending on how close you are to that line. I would much prefer a straight line non-risk averse thing. I think Tom had talked about before.

I also thought, and I'm not sure I got the answer right, I looked at the effect of the dropping and SSC to see what its effect was on both risk and consequence. I got the impression that you looked at those SSCs one at a time.

I had the same problem that John had. It depends on what order you do the safety function as to what kind of answer you would get. Plus, I see that maybe one SSC puts you into a different frequency category or consequence.

But if you did two of them even if they're not in the same safety function if you're separating down the line on the thing then those two together might put you in the unacceptable regions of a top-level criteria. And I was wondering why -- those being the safety-related. But they're relatively independent. Maybe that's the issue.

So, the one thing that I keep harping on every time because it's an issue I've had back when I was on the ACRS was we're still using the prompt fatality safety goal as our risk acceptance criteria for all the -- building all the PRA events.

I'm still saying that's not the control room, it's the societal risk of total deaths and all the dollars. And I'm glad to hear Tom mention that. That has been looked at to some extent. But we don't have any acceptance criteria for societal risk to compare it to. Or we don't -- maybe we need some.

I think this is appropriate for these new reactors because even though meeting the PAGs at the EAB probably means they're going to meet the two QHOs very well, but you may not meet the societal risk because we don't know what it is and it's bigger and doesn't take as much release to get a pretty good amount. So that's been one of my issues. I think we need to think about the total effects of the release.

I think there's going to be a lot of work to -- and I think some of it's going on to get good fission product release in transport models, particularly issues with the plateout and re-vaporization from primary systems, and effects of dust. How do we know how much dust is going to be there.

And I think there's still some issues on how you address air ingress. Every time I talked about gas-cooled reactors when I used to work, that's before I quit working, I always asked how about dropping a vein with air in there. You know, we had Chernobyl with all that burning going on. I don't think burning of this graphite with air ingress is a big issue but I think it needs to be put to bed and say oh, we're really not going to have that kind of frequency of air ingress or I don't think you're going to have -- with this kind of graphite I don't think you're going to have so much of a problem with the burning of it.

But I do think you need to address the issue. You need to know what the effects of air ingress and the graphite interaction are. I'm not sure I saw that in the proposed research program.

And I'd also think that -- I would support one of these operational processes where you raise the power up in steps. Because I'm worried about -- I mean the fuel quality has to be so good that it doesn't take much of a mistake to not have the fuel quality you think.

I think one of the things you're going to need to do is be able to assess the fuel quality as you operate. And I was glad to hear there's intentions to putting in instrumentation and measuring reactivity, plateout and airborne, in the RCCS as you go along.

But I think, I haven't seen any criteria yet on what would you do when you start getting too much activity, and what is too much, and when do you decide on when to shut down. I would like to see more on that.

And I think the issue with PRA being not quite as mature as we'd like is an issue. I think you address it with trying to determine what the uncertainties are. And I don't know if we have programs or ways to get the uncertainties. You know, you look at all the safety functions and the failure of SSCs and how they lead to the final product in both frequency -- frequency is easier to get consequences. But I think you need a lot of data to get a mature PRA.

Let's see. And I don't know how to -- I think to do with modular reactors my feeling is it's probably sufficient to use consequences for one module and just deal with the frequencies for multiple modules. But I haven't thought through the issues of simultaneous initiating events. Even those I think it's probably all right to do it that way but maybe not. I haven't thought that one through yet.

But I think we need to do a little more thinking about what actually defense-in-depth is, how we define it. When is it we have sufficient -- how do we deal with it in terms of uncertainties and confidence levels. And how we deal with it in terms of redundancy and that sort of stuff.

In general, you know, in spite of the fact that I'm throwing up things like this I like what I've heard. I think the staff's doing a good job. I think that the applicant or whatever, the DOE-INL people, I think it's a good piece of work. I'm glad to see it going on.

CHAIRMAN BLEY: Thank you. And I just have a few comments. I said most of the things I wanted to during the session.

I dwelled a lot on an issue of the deterministic DBAs. And I think that's probably not a big issue because to my thinking when they go through building the best PRA they can with all the scenarios they can think of. They use the same kind of process you're talking about. If one generates new sequences they didn't think about they must get reflected back into the PRA and get evaluated.

In the end I think the format they laid out, the framework for defense-in-depth is perhaps more thorough than most I've seen and gives a good way to require additional protection in areas where we're not completely confident on the uncertainties that have been evaluated or any of the technology issues that we'd hope to get to see through the prototype.

We keep hearing credible events, and we're only going to look at credible events, we look at credible events. It's an ill-defined term and it gets us into trouble every time. You look at things that are physically possible and once you do that you either take care of them because you're not sure if they can occur or not, or you look at them probabilistically and evaluate them. I just get uncomfortable with that phrase tossed around so much. It's really ill-defined.

In the same vein I don't think a deterministic look solves our problems. I mean, where we started way back when was we had a bunch of smart guys think up everything they could think of that would happen as judgment. Well, we've now got a bunch of guys trying to think up everybody they can account for and maybe analyze them a little more thoroughly.

But I don't see those as separate things. You need that same process to come up with the events. If you come up with more than people thought of you incorporate that in the process.

The one interesting thing about what John brought up, and I haven't gone back and looked carefully enough because I didn't notice it, and Tom did too, this idea that when you change the ordering of the things in the event trees and you get into a different category so that what you've really got is a model that's not coherent for some reason. And there are lots of reasons that can happen. So I think it would behoove everyone to go back and look at that and see what's going on and find a way to account for it.

It needs to be done and there's a whole source of technical issues that can be involved in that and they range from shadowing of one thing with the other to incorporating the effects of one and the other. And if you think of it as conditional, you have to look at that whole set as conditional and make sure we get it right.

Having a system that lets you come out with different categorizations depending on the whim of the analyst and how they break out their model or how they order it is something you've got to find a way around. That could be a really significant problem.

And I'd throw out the PRAs when they finally come in for a real design need to incorporate all those external events, the whole group of them. But they also need to incorporate human actions including what's historically been called errors of commission. But the kind of things you brought up, turning on the circulators.

If that's a big deal that needs to be in the model as a possible thing that could happen. And it eventually gets worked into the training and everything else to make it much less likely that they take the wrong action.

But you can't just look at operators doing what you expect them to do. You have to look at what they might do to get us into trouble. And the thinking process you talked through is what has to be there and has to be in the PRA as well.

Anyway, thanks everyone. I was very impressed with the presentations and the discussions and how this went on. I don't know what happens in the future. We're interested in hearing how the fuel work continues. I wish more were continuing in other areas but I guess it won't for some time.

Thanks everyone and my compliments to you. And at this point the meeting is adjourned.

(Whereupon, the foregoing matter went off the record at 5:16 p.m.)

U.S. DEPARTMENT OF ENERGY | Nuclear Energy

ACRS Future Plant Designs Subcommittee Meeting

NGNP Introduction

Carl J. Sink
NGNP Program Manager
Office of Nuclear Energy
U.S. Department of Energy

April 9, 2013

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

Continued DOE Focus on Licensing Framework

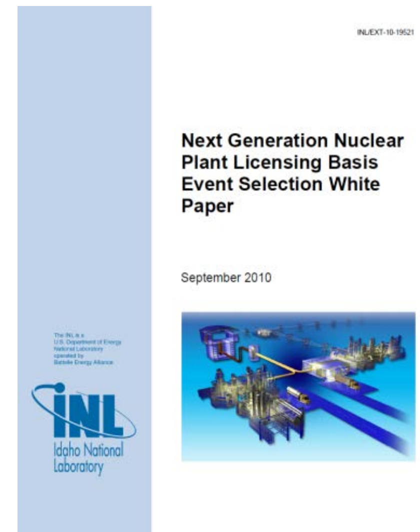
Secretary Chu letter to Congress in October, 2011 reinforces the priority that DOE places on establishing the HTGR licensing framework, based on the related NEAC recommendation

- "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

DOE appreciates NRC's support of the significant level of interaction that has taken place within the NGNP program for licensing framework development:

- Jointly established Licensing Strategy for adaptation of existing regulations
- Review and feedback on NGNP white papers covering various licensing framework topics
- Significant number of public meetings (18 total) over the past 3 years
- Review of the NGNP responses to approx. 450 requests for additional information
- Review of technology development plans
- Approval of the applicable portions of the NGNP Quality Assurance Program Description
- Feedback on the highest priority licensing issues, as described in NGNP's July 6, 2012 letter



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 dated February 15, 2012

- **NRC staff positions have been requested in four key areas**
  - Licensing Basis Event Selection
  - Establishing Mechanistic Source Terms
  - Functional Containment Performance Requirements
  - Development of Emergency Planning and Emergency Planning Zone Distances

U.S. DEPARTMENT OF ENERGY

Nuclear Energy

Reducing Regulatory Uncertainty for HTGRs

- **DOE is focused on the resolution of long-standing HTGR licensability issues, and the establishment of key parts of the NGNP licensing framework**

- **The proposed NGNP framework provides a process for assuring, with associated fuel qualification program results to date, adequate protection of the public over a wide spectrum of internal and external events at a multi-reactor module plant facility, with significant margin to the regulatory requirements for offsite dose**

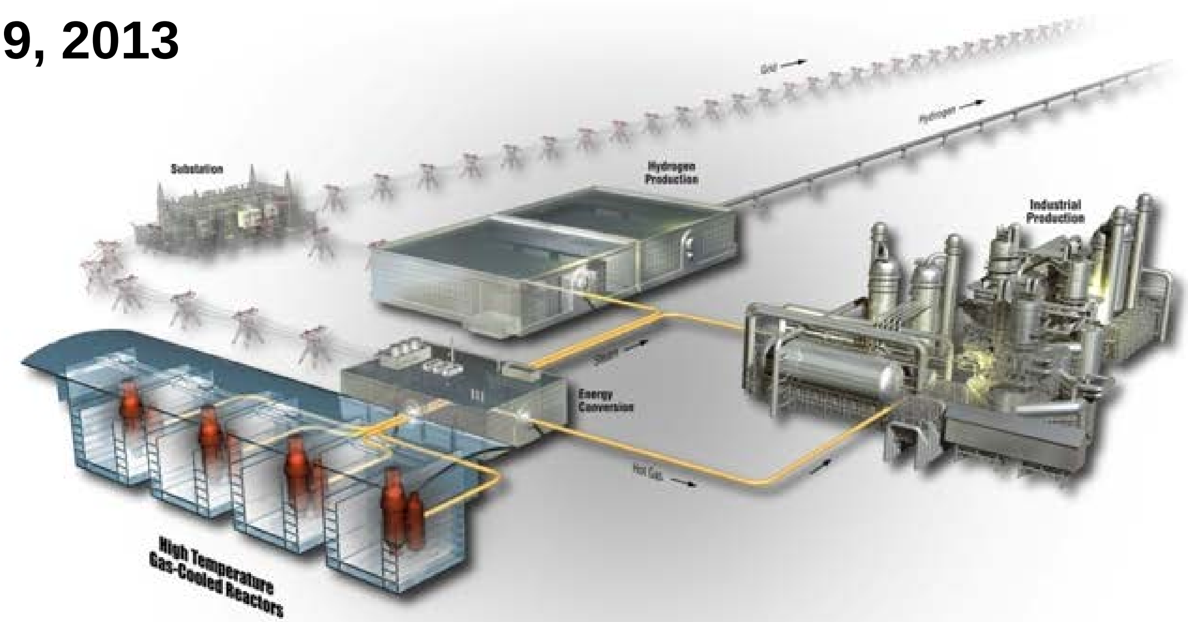
- **DOE looks forward to today's follow-on meeting regarding the most significant topics affecting the licensing framework for NGNP**

NGNP
Next Generation Nuclear Plant

Summary of January 17 NGNP Presentations

ACRS Future Plant Designs Subcommittee Meeting

April 9, 2013



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Idaho National Laboratory

NGNP January 17 Presentations to the Subcommittee Addressed Five Areas

- High Temperature Gas-Cooled Reactor (HTGR) Safety Approach and Design Basis
- Licensing Basis Event (LBE) Selection Process
- Functional Containment and Mechanistic Source Terms
- Siting Source Terms (SST)
- Fuel Qualification and Radionuclide Retention

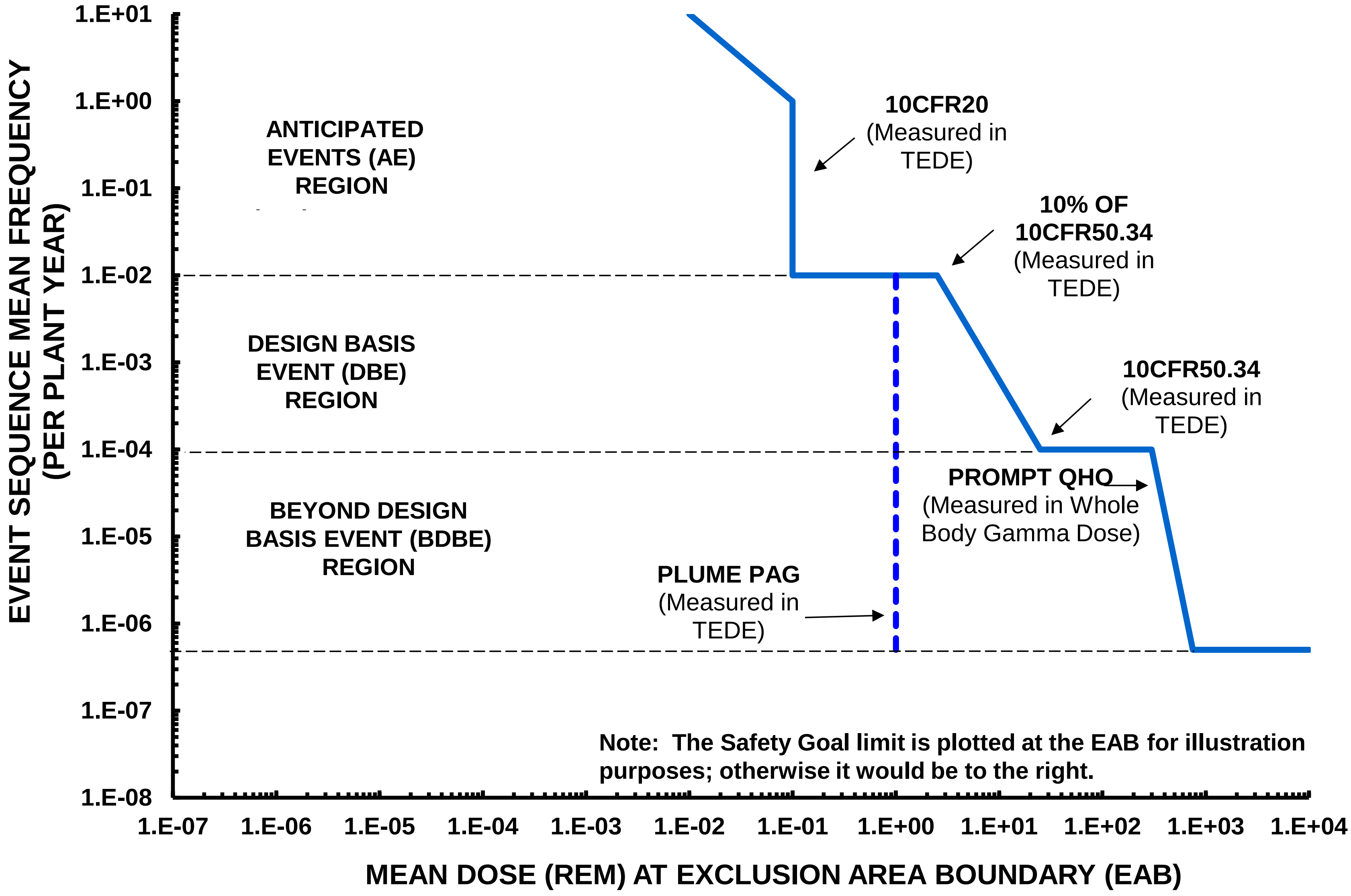
Safety Approach and Design Basis Summary

- Top objective is to meet the NRC offsite dose requirements and EPA Protective Action Guides (PAGs) at the Exclusion Area Boundary (EAB) for spectrum of events within and beyond the design basis
- Responsive to Advanced Reactor Policy
- Modular HTGR designs employ multiple concentric, independent barriers to meet radionuclide retention requirements – these barriers comprise the Functional Containment
  - Fuel Elements
    - Fuel kernels
    - Particle coatings (most important barrier)
    - Compact matrix and fuel element graphite
  - Helium Pressure Boundary
  - Reactor Building
- Emphasis is on radionuclide retention at the source within the TRISO fuel coatings
  - Passive heat removal
  - Control of heat generation
  - Control of chemical attack

LBE Selection Summary

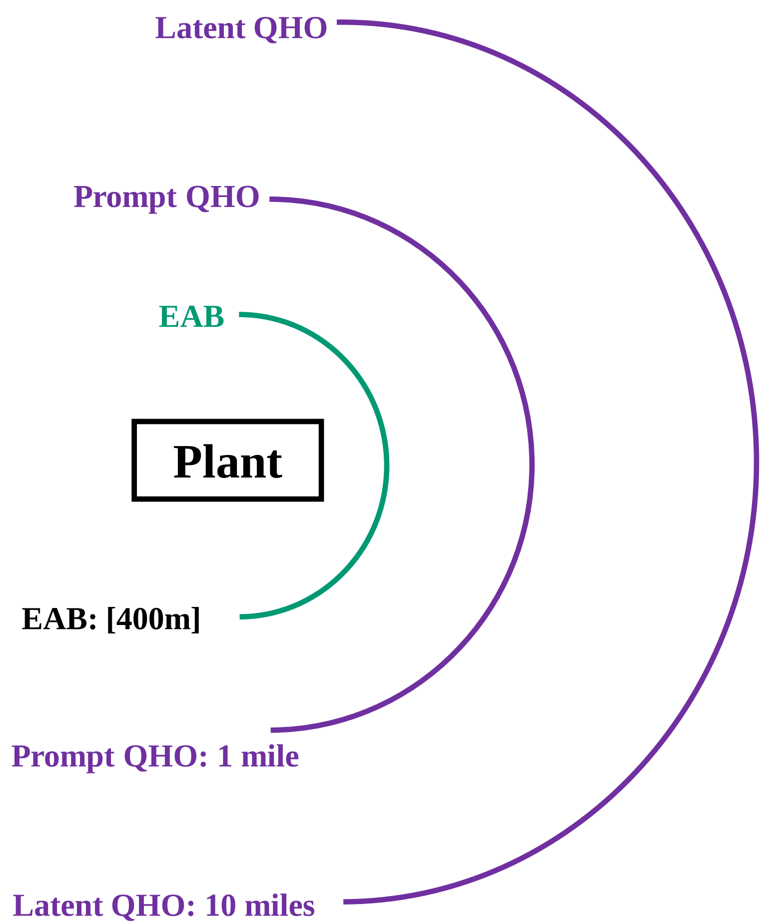
- Licensing Basis Events determine when Top Level Regulatory Criteria (TLRC) must be met
- Selected during design and licensing process with risk insights from comprehensive full scope PRA that considers uncertainties
- Include anticipated events (AEs) (expected in life of plant), design basis events (DBEs) (not expected in plant lifetime), beyond design basis event (BDBEs) (not expected in fleet of plant lifetimes), and design basis accidents (DBAs) (Ch 15 events derived from DBEs with only safety related structures, systems, and components [SSCs] available)
- Safety classification focuses on examining SSCs available and sufficient to successfully perform required safety functions to mitigate spectrum of DBAs

NGNP Frequency-Consequence Curve



Selection of TEDE and EAB for TLRC Dose Limits

- Mean total effective dose equivalent (TEDE) selected for consequence measure
- The 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, with the goal of having the LPZ and EPZs at the same distance as the EAB (approximately 400m)
  - If met, the plant will have large margins to the average individual risk Quantitative Health Objective (QHOs) as measured within annular regions from the EAB to 1 and 10 miles, respectively
  - Supports co-location with industrial facilities



Licensing Basis Event 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

Design Basis Accident Derivation and Dose Limits

- DBAs (analyzed in Chapter 15 of SARs) are deterministically derived from DBEs by assuming that only safety-related SSCs are available
- The event sequence frequency for some DBAs is expected to fall in or below the BDBE region
- Consistent with traditional practice, DBAs must meet the DBE dose limits based on conservative (upper 95%) analyses, including those with event sequence frequencies in or below the BDBE region
- DBAs are not derived from BDBEs. BDBEs must meet the NRC QHO on a cumulative basis based on an expected (mean) analysis

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Performance Standard for Functional Containment

NGNP's upper tier performance standard for the functional containment ensures the integrity of the fuel particle barriers rather than allowing significant fuel particle failures and then relying extensively on other mechanistic barriers (e.g., the helium pressure boundary and the reactor building). This standard is characterized by the following:

- Ensure radionuclide retention within fuel during normal operation with relatively low inventory released into the helium pressure boundary.
- Limit radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria at the EAB with margin for a wide spectrum of off-normal events.
- Maintain the capability to establish controlled leakage and controlled release of delayed accident source term radionuclides.

Functional Containment Performance Summary

- Radionuclide retention within fuel during normal operation with relatively low inventory released to helium pressure boundary (HPB)
- Limiting LBEs 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 the wide spectrum of DBEs and DBAs 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

Functional Containment and Mechanistic Source Terms Approach Summary

- Mechanistic models of fission product generation and transport that account for reactor inherent and passive design features and the performance of the radionuclide barriers that comprise the functional containment
- Event specific and applied to the full range of licensing basis events affecting one or more modules
- Consistent with the NRC Advanced Reactor Policy Statement
- 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

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Siting Source Term Summary

- The NGNP approach to SSTs 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
- The approach implements a modular HTGR-appropriate interpretation of the 10CFR50.34 (10CFR52.79) footnote regarding siting evaluation
- Limiting DBAs are evaluated to determine SSTs
- Further, to ensure that there are no cliff edge effects, physically plausible Bounding Event Sequences (with frequencies below the BDBE region), including those involving graphite oxidation, are considered

Fuel Qualification and Radionuclide Retention Summary

- The Fuel Development and Qualification Program is providing data, under an NRC-accepted QA program, necessary to better understand fuel performance and fission product behavior for modular HTGRs

- The Fuel Program is laying the technical foundation needed to qualify UCO TRISO fuel made to fabrication process and product specifications within an envelope of operating and accident conditions that are expected to be bounding for modular HTGRs

- Results to date are consistent with current design assumptions about fuel performance and radionuclide retention. The program is obtaining additional data to support model development and validation

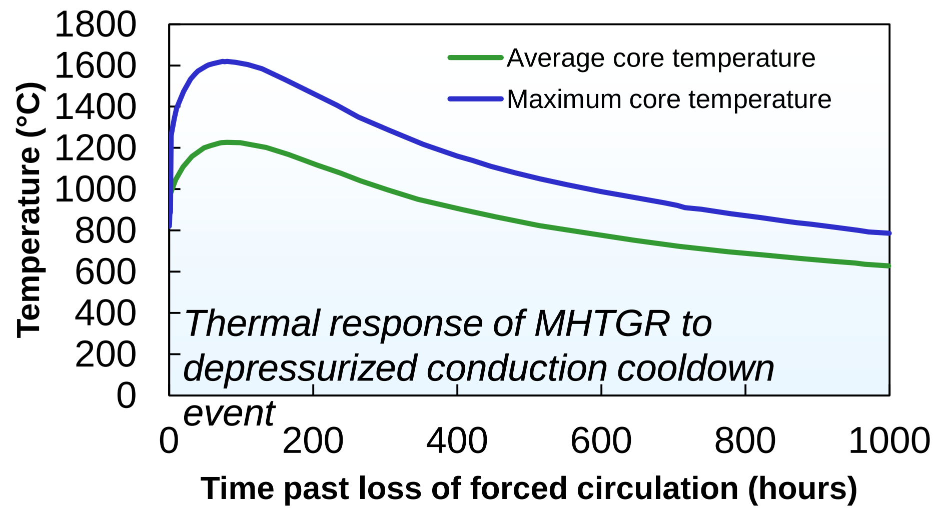
- Results to date support the safety design basis, including the functional containment and mechanistic source term approaches

Key Results of On-going Fuel Research

- Improved understanding of TRISO fuel fabrication process
- Improved fabrication and characterization of TRISO fuel produced by fuel vendor
- 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 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 for hundreds of hours at 1600, 1700, and 1800°C demonstrating robustness of UCO TRISO under depressurized conduction cooldown conditions

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
- Six isothermal 1600, 1700, and 1800°C tests have been completed
- Actual time-temperature test to be performed this year
- Testing of deconsolidated particles will occur in late 2013 or early 2014



KEY RESULTS

- Releases not seen from the intact TRISO particles during the high temperature heating
- Releases that have occurred are very low and are from one or more of the following:
    - Fission products that diffused into the matrix during irradiation
    - Presence of a defective particle
    - A particle that fails during safety testing

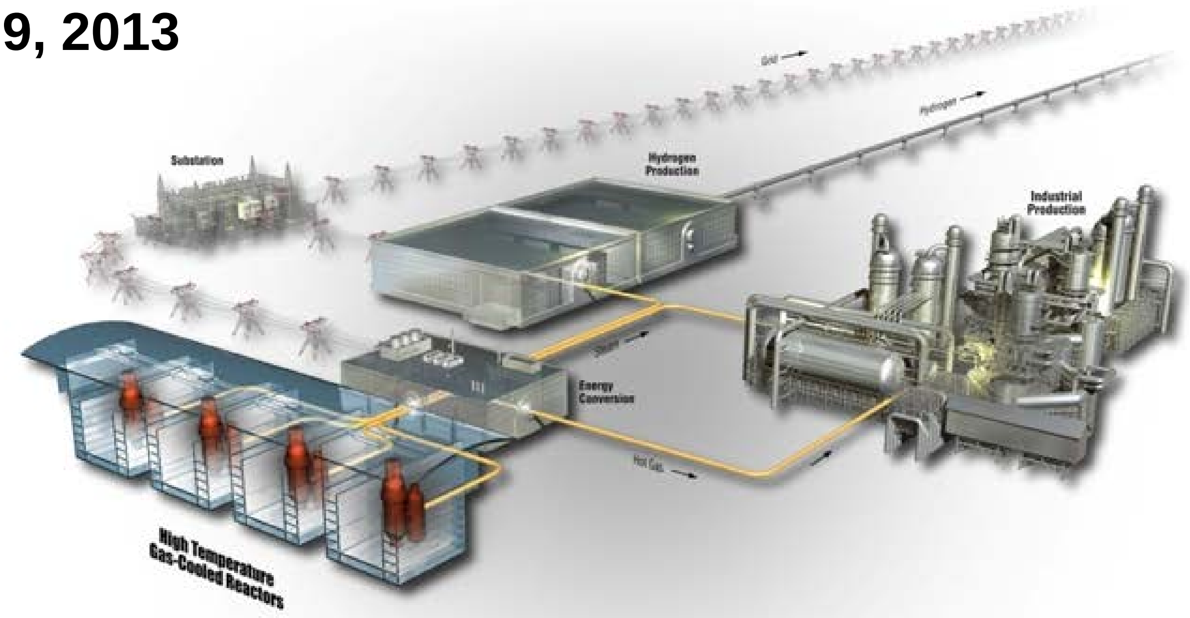
The Subcommittee Asked Questions in Two Areas to be Addressed at Today's Meeting

- What is the role of the reactor building in defense-in-depth?
- What is NGNP's approach to defense-in-depth?

Reactor Building Design Alternatives

ACRS Future Plant Designs Subcommittee Meeting

April 9, 2013



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INL Idaho National Laboratory

Key Modular HTGR Safety Attributes

- The fuel, helium coolant, and graphite moderator are chemically compatible under all conditions
- The fuel has very large temperature margins in normal operation and during accident conditions
- Safety is not dependent on maintaining the helium coolant pressure, and loss of coolant pressure does not transfer large amounts of energy into the reactor building
- Post accident heat removal is accomplished by passive means
- Response times of the reactor are very long (days as opposed to seconds or minutes)
- The HTGR has multiple, concentric, independent radionuclide barriers. A breach of the helium pressure boundary does not result in failure of the fuel or the reactor building

Role of the Reactor Building (RB) in Safety Design

- Required safety function of the RB is to provide structural protection, from internal and external events and hazards, for passive heat removal from Reactor Vessel to Reactor Cavity Cooling System (RCCS)
  - Maintain Vessel System/Helium Pressure Boundary (HPB) geometry
  - Maintain RCCS geometry

- The RB provides other functions not required to meet regulatory requirements for offsite dose
  - Provides additional radionuclide retention (needed to meet EPA PAGs at EAB)
  - Limits air available for ingress after HPB depressurization

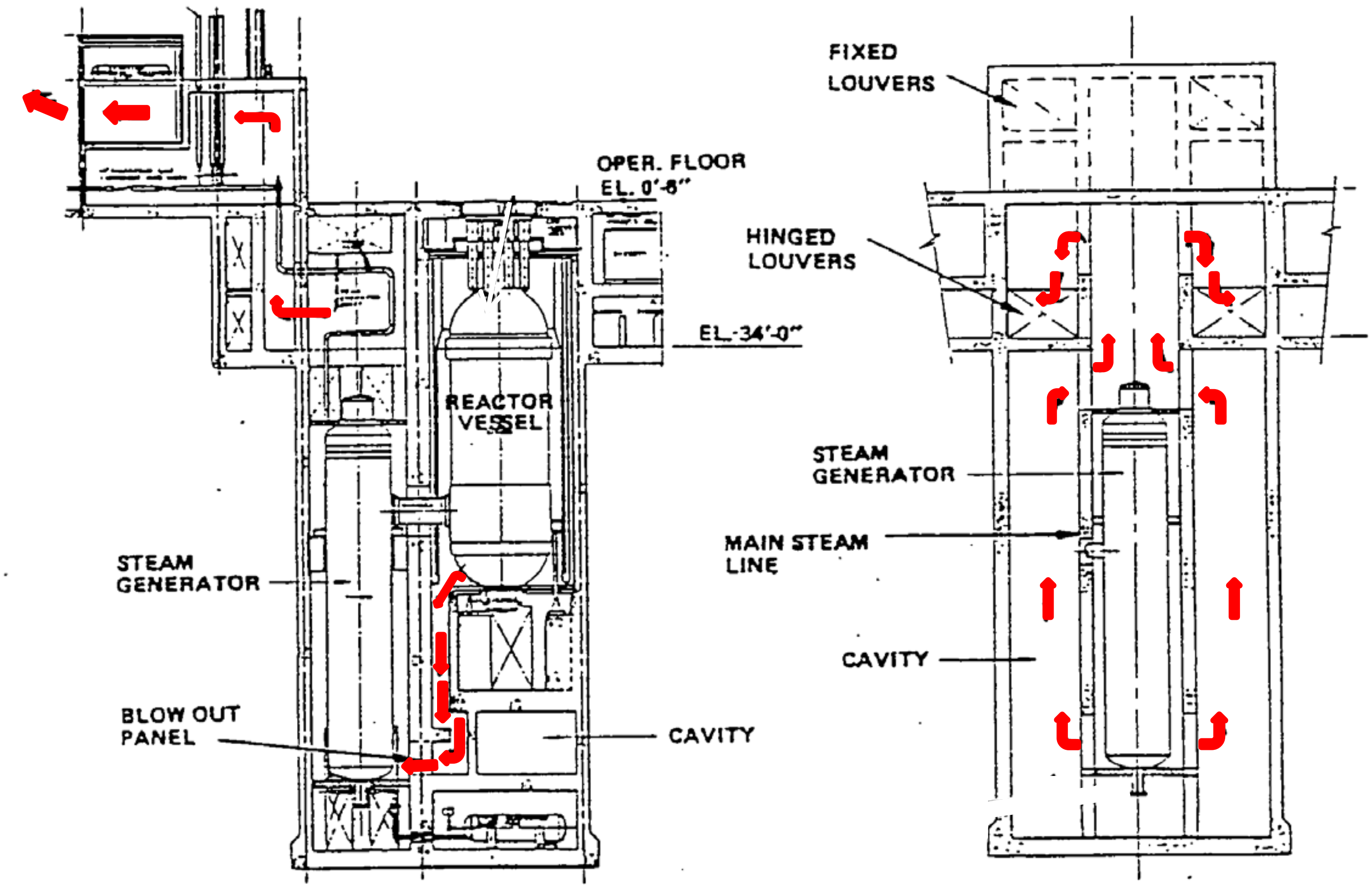
Vented Reactor Building Addresses Several Modular HTGR Specific Design Issues

- Compatible with non-condensing helium coolant

- Matched to modular HTGR accident behavior
  - Vented early in transient when radionuclides released from helium pressure boundary are relatively low
  - Closed later in transient when radionuclides released from fuel are relatively high

- Provides a more benign environment (e.g., heat, pressure, and structural loads) for passive Reactor Cavity Cooling System

MHTGR Reactor Building Vent Path from Reactor or Steam Generator Cavities



*Alternative RBs Considered in Containment Study for MHTGR**

1. Vented, moderate leakage (100%/day) (Reference)
2. Vented, filtered, moderate leakage (100%/day)
3. Vented, filtered, low leakage (5%/day)

4A. Unvented, moderate pressure, low leakage (5%/day) air RCCS

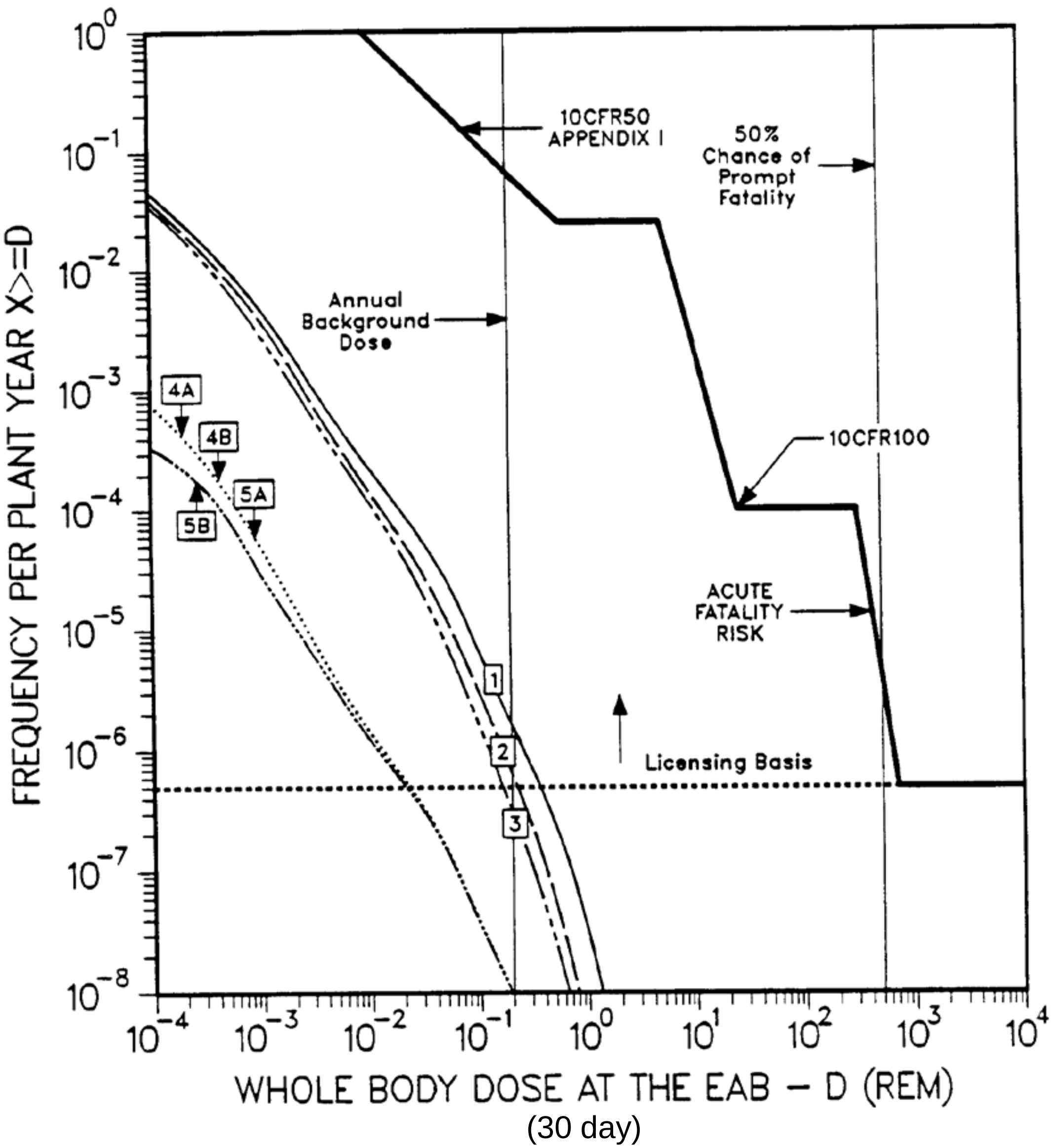
4B. Unvented, moderate pressure, low leakage (5%/day) water RCCS

5A. Unvented, low pressure, low leakage (5%/day)

5B. Unvented, low pressure, low leakage (1%/day)

* "Containment Study for MHTGR," General Atomics Report, DOE-HTGR-88311, November 1989

All Alternative Reactor Buildings Considered for MHTGR Met the TLRC with Substantial Margin



Alternative PBMR RB Design Configurations

1a. Unfiltered, vented, moderate leakage (50-100%/day)

1b. Unfiltered, vented with blowout panels, moderate leakage (50-100%/day)

2. Partially filtered, vented with blowout panels, moderate leakage (50-100%/day)

3A. Filtered, vented with blowout panels, lower leakage (25-50%/day)

3B. Filtered, vented with blowout panels and expansion volume, lower leakage (25-50%/day)

4A. Pressure retaining with internal blowout panels, low leakage (<1%/day)

4B. Pressure retaining with internal blowout panels and expansion volume, low leakage (<1%/day)

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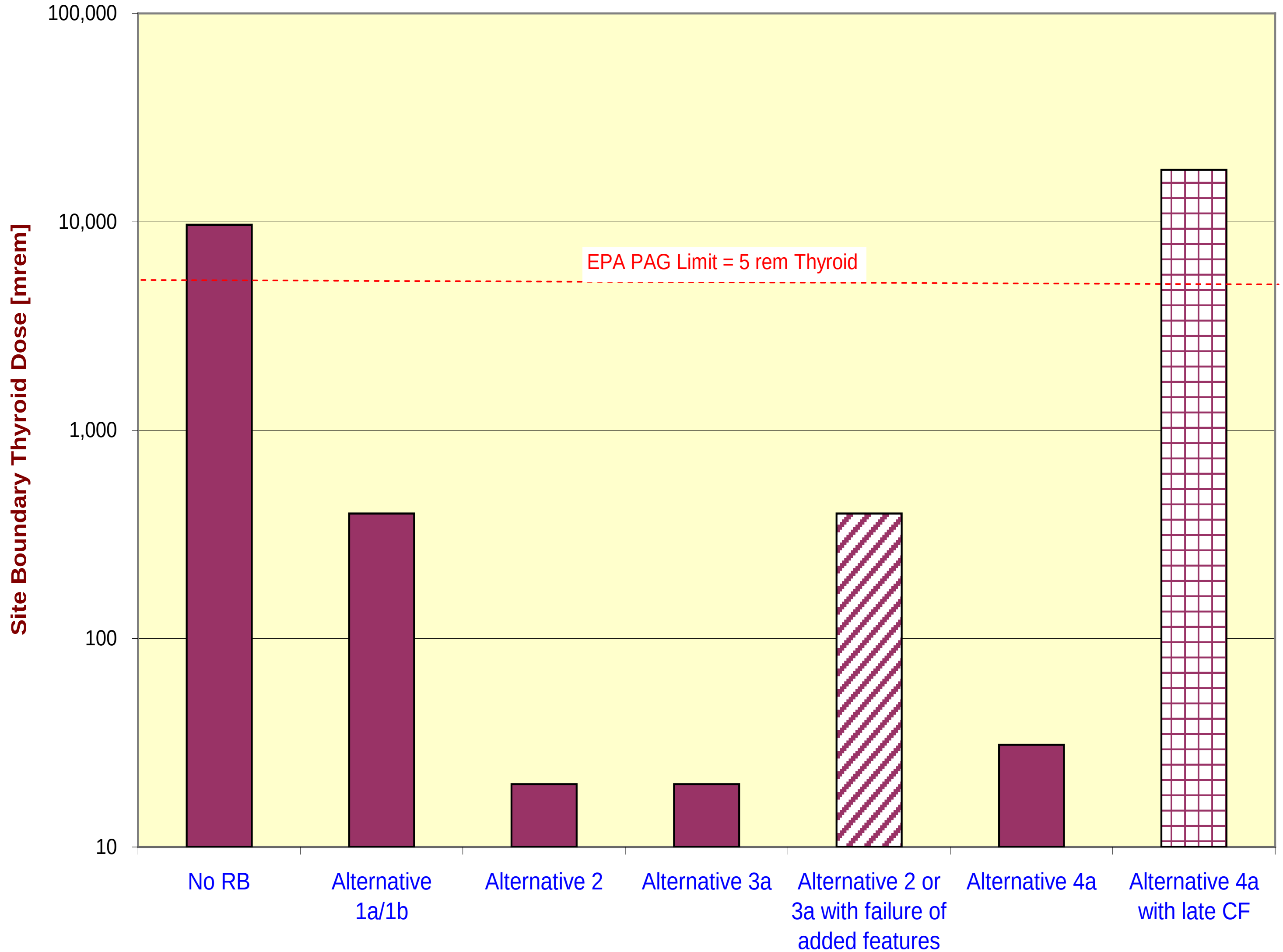
* “Reactor Building Functional and Technical Requirements and Evaluation of Reactor Embedment,” NGNP-NHS 100-RXBLDG, Rev 0, Westinghouse PBMR Team Report, September, 2008.

PBMR Reactor Building Alternatives 1a thru 4a Met the EPA PAG at the EAB with Substantial Margin



Summary of Findings from MHTGR and PBMR Alternative RB Evaluations

- Vented building provides best match for modular HTGR characteristics and passive design
- For modular HTGRs, high pressure, low leakage LWR-type containment designs increase radionuclide release in low frequency events
- Added filters and/or active HVAC systems that may not be available for low frequency events (e.g., seismic or station blackout) provide little additional margin relative to the TLRC
- Confirmed decision to place emphasis on retention at the source within the fuel
- More detail can be found in the response to RAI FQ/MST-82 and in its references

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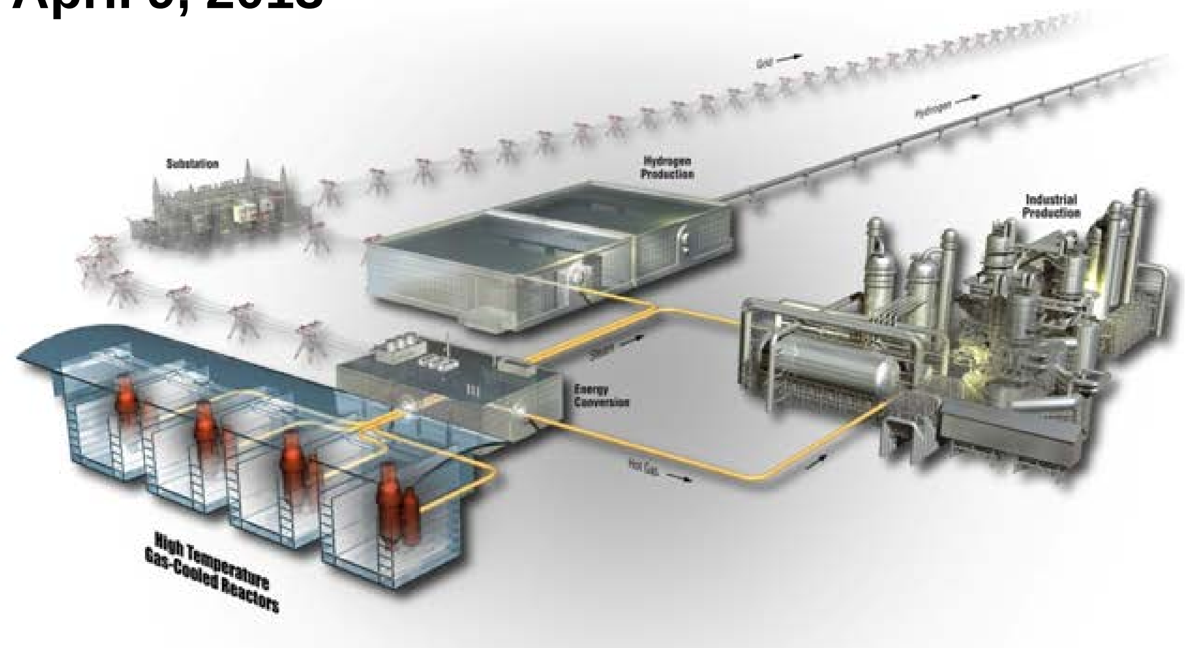
NGNP Defense-in-Depth Approach

ACRS Future Plant Designs Subcommittee Meeting

April 9, 2013



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Idaho National Laboratory

Presentation Agenda

- Defense-in-Depth (DID) Overview
  - DID Approach
  - DID Elements
  - NRC's DID Strategy
- NGNP DID Approach
  - Plant Capability DID
  - Programmatic DID
  - Risk-Informed Evaluation of DID
- Integrated DID Framework
- NGNP DID Approach Summary
- Key NGNP Attributes

DID Approach

- Develop a structured system for evaluating DID adequacy for licensing

DID Elements

- Plant Capability DID
- Programmatic DID
- Risk-Informed Evaluation of DID

Recent Summary of NRC's DID Strategy

"To protect public health and safety from the inadvertent release of radioactive materials, the NRC's defense-in-depth strategy includes multiple layers of protection:

(1) prevention of accidents by virtue of the design, construction and operation of the plant,
(2) mitigation features to prevent radioactive releases should an accident occur, and
(3) emergency preparedness programs that include measures such as sheltering and evacuation.

The defense-in-depth strategy also provides for multiple physical barriers to contain the radioactive materials in the event of an accident."

EA-12-050, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents," Page 6, March 12, 2012.

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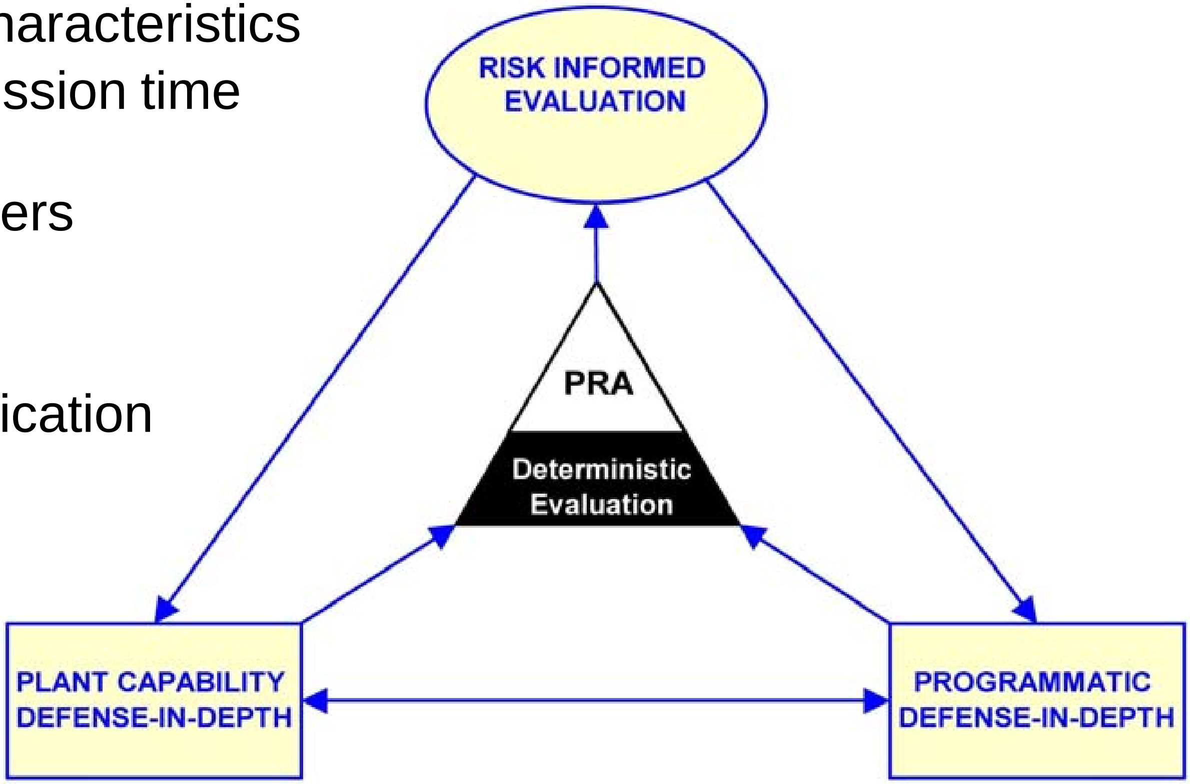
Plant Capability DID

- Reflects the decisions made by the designer in the selection of functions, structures, systems and components (SSC) for the design that assure defense-in-depth in the physical plant
- Examples:
  - Inherent reactor characteristics
  - Long event progression time constants
  - Radionuclide barriers
  - Passive SSCs
  - Active SSCs
  - SSC safety classification
  - Design margins



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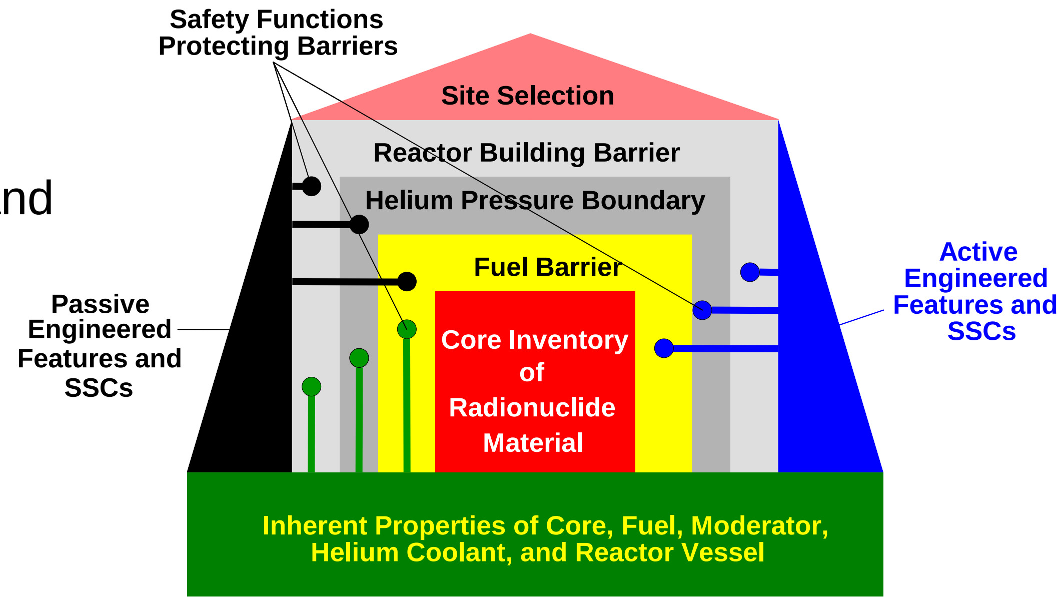
Radionuclide Barriers

Use multiple barriers for radionuclide retention

- The radionuclide barriers are concentric and independent
- Emphasis is on the performance of the fuel barriers
- Reactor Building provides DID for meeting top level regulatory criteria (TLRC)
- Active (typically non-safety related) SSCs and passive (typically safety related) SSCs work in concert with the inherent design characteristics to reduce the frequency of challenges to radionuclide barriers
- Challenges to barrier integrity and independence are considered
- Safety margins and conservative design approaches are used to address uncertainties in barrier and SSC performance



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Control of Core Heat Generation

- 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 reactor subcritical
- Either system capable of maintaining cold shutdown during refueling

Removal of Core Heat

- Heat Transport System (active)
  - Used for process steam/cogeneration during normal operations
  - Residual heat removal with forced cooling of pressurized or depressurized helium from core to steam generator to secondary heat sink
- Shutdown Cooling System (active)
  - Provides heat removal during planned maintenance and unplanned events for investment protection
  - Residual heat removal with forced cooling of pressurized or depressurized helium from core to shutdown cooling water system
- Reactor Cavity Cooling System (passive)
  - Provides heat removal for investment and public protection during off-normal events
  - Residual heat removal from low power density, high heat capacity annular core with convection, conduction, and radiation to reactor vessel with helium pressurized or depressurized
  - Radiation from uninsulated reactor vessel to natural convection system in reactor cavity (air or water)

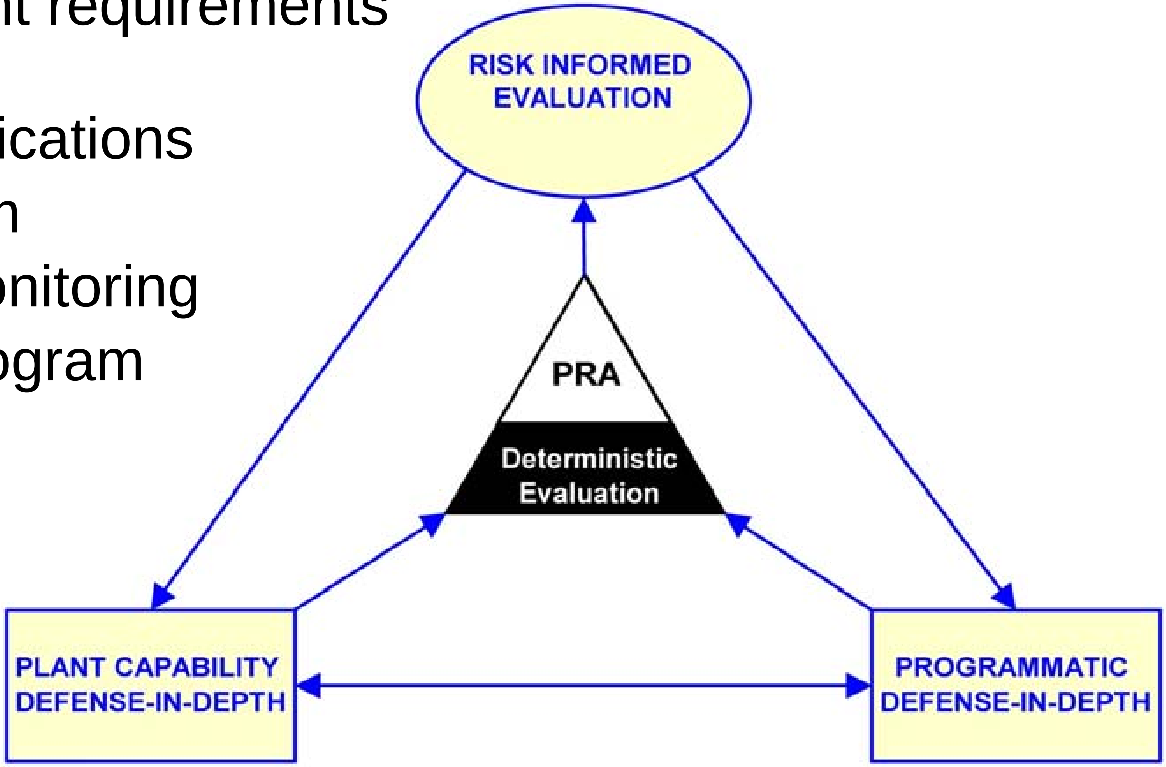
Control of Chemical Attack

- Air Ingress
  - Non-reacting coolant (helium)
  - High integrity nuclear grade pressure vessels make a large break exceedingly unlikely
  - Slow oxidation rate (high purity nuclear grade graphite)
  - Limited by core flow area and friction losses
  - Reactor building embedment and vents that close after venting limit potential air ingress
- Water Ingress
  - Non-reacting coolant (helium)
  - Limited sources of water with moisture monitors, steam generator isolation (does not require AC power) and steam generator dump system
  - Water-graphite reaction is endothermic, requires temperatures > normal operation, and has a slow reaction rate
- Graphite fuel element, fuel compact matrix, and ceramic coatings protect fuel particles

Next Generation Nuclear Plant

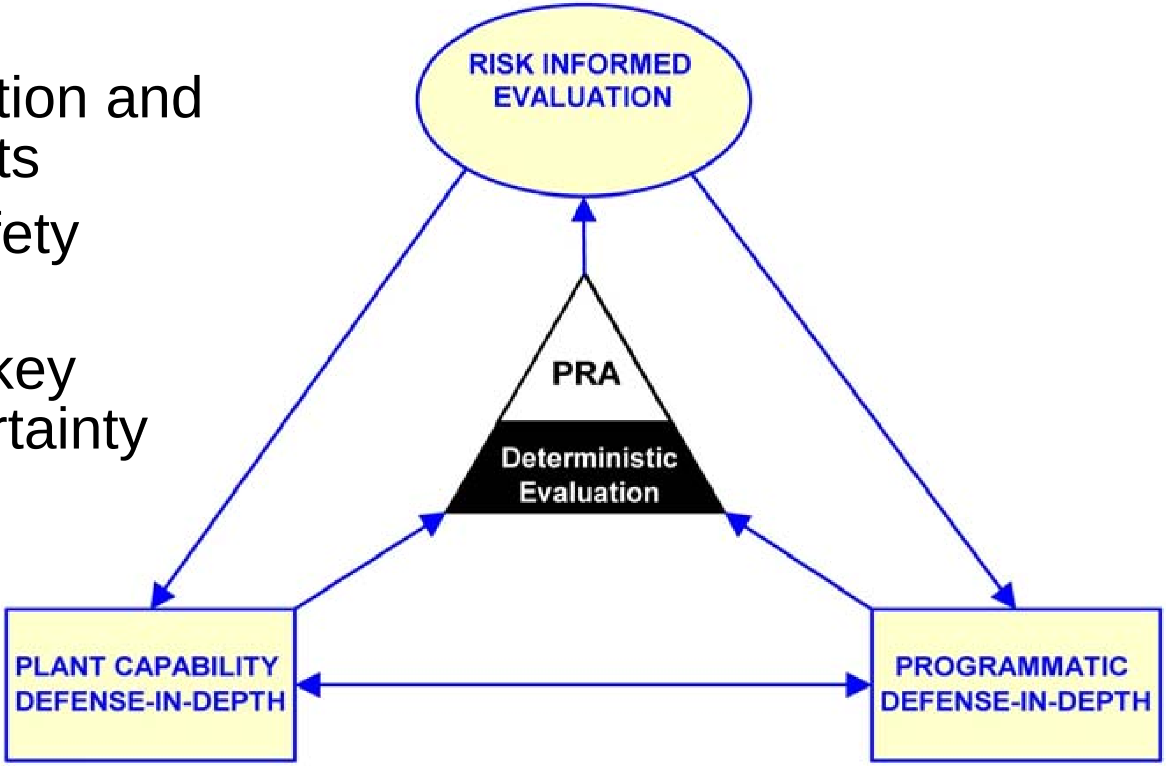
Programmatic DID

- Processes of manufacturing, constructing, operating, maintaining, testing, and inspecting the plant that assure plant safety throughout the lifetime of the plant
- Examples:
  - Special treatment requirements for SSCs
  - Technical specifications
  - Training program
  - Performance monitoring
  - Maintenance program



Risk-Informed Evaluation of DID

- Provides the framework for performing deterministic safety evaluations and risk assessment evaluations to determine how well various Plant Capability and Programmatic DID strategies have been implemented
- Provides:
  - Accident prevention and mitigation insights
  - Input to SSC safety classification
  - Identification of key sources of uncertainty



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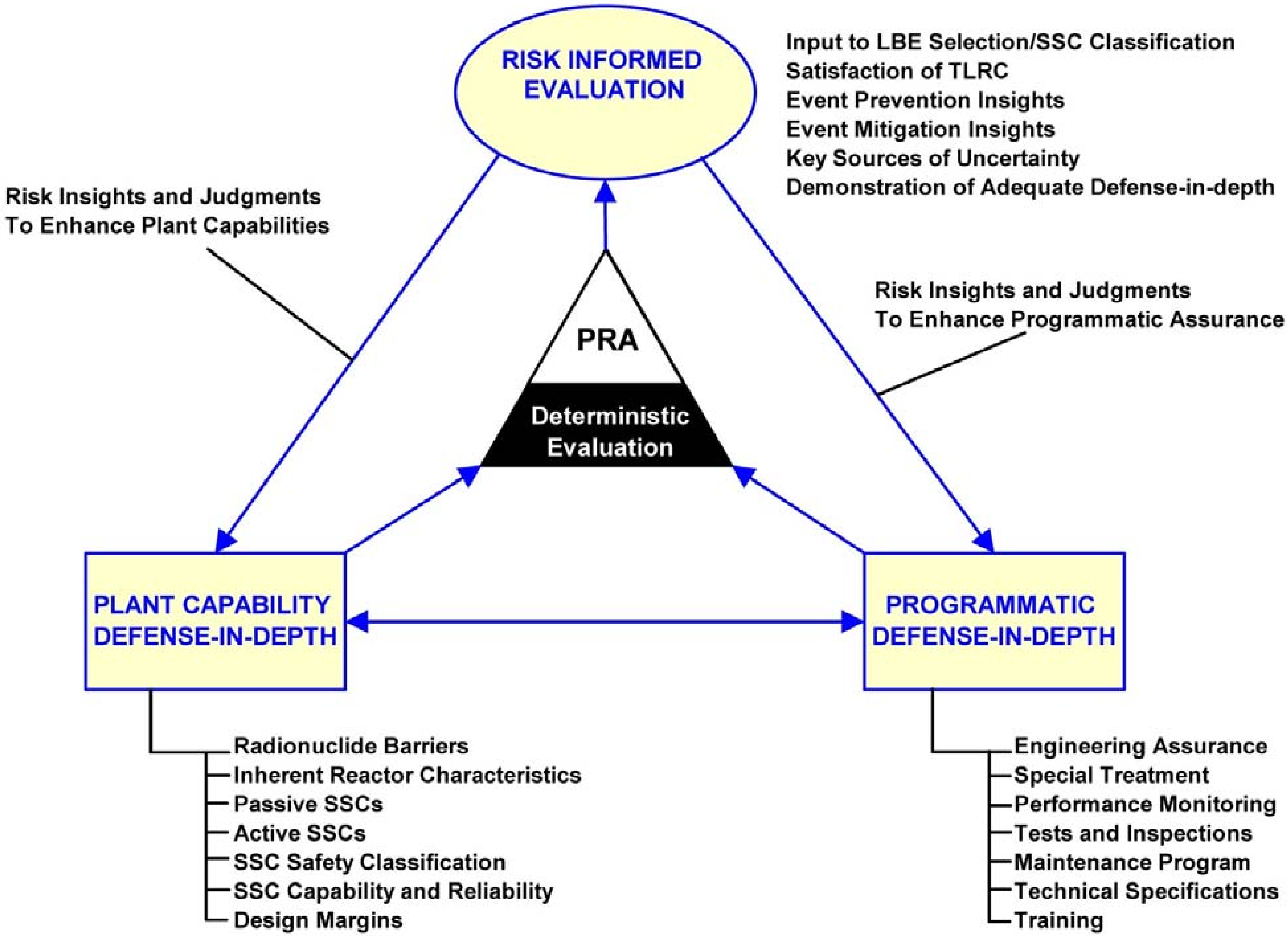
Risk-Informed Evaluation of DID

- Identify credible failure modes and challenges to the radionuclide barriers; include dependencies and interactions among barrier and other SSC failure modes
- Identify the roles of SSCs in the prevention and mitigation of accident sequences and quantify the extent to which the accidents are prevented and mitigated
- Establish that there are no events with a significant frequency of occurrence that rely on a single element of design or programmatic approach in protecting the public from a release whose dose would exceed the TLRC

Integrated DID Framework



NGNP DID Approach Summary

Evaluates plant design capability features and programmatic elements in an integrated risk management approach to identify opportunities to reduce risk and to ensure that an adequate treatment of DID has been achieved after considering a full spectrum of events

Prevention

- Ceramic fuel resistance to melting
- Long event progression time constants
- Low power density; high heat capacity; slender, annular core geometry for heat transfer

Mitigation

- Multiple (independent) barrier approach against radionuclide releases
- Active SSCs and passive SSCs work in concert with the reactor's inherent characteristics to protect the public
- Addresses uncertainty by employing safety margins and special treatments to ensure SSC capability and reliability

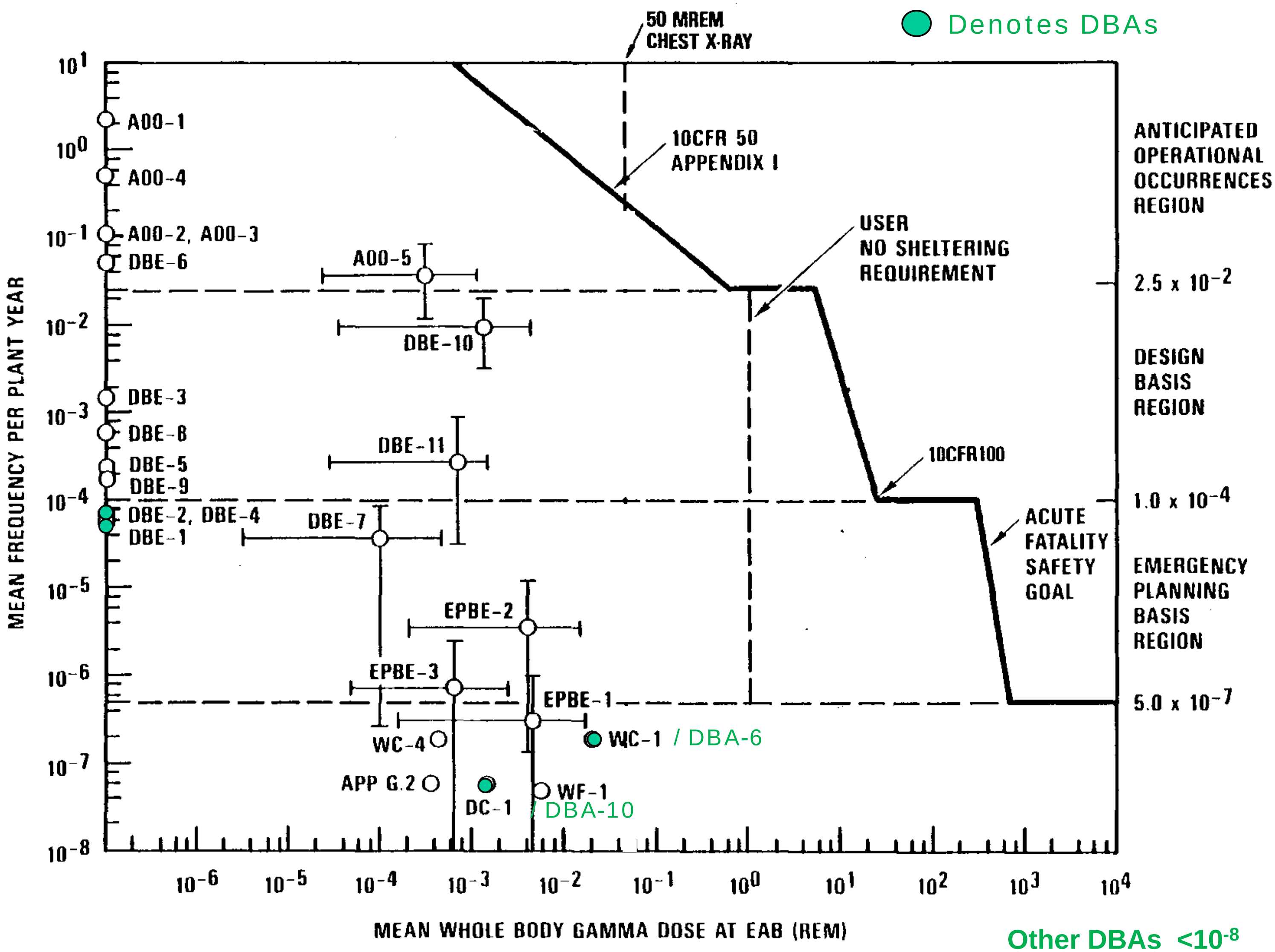
Emergency Preparedness

- Design goal to meet the PAGs at the site boundary (EPZ) for DBEs and BDBEs provides margin to the TLRC

Key NGNP Attributes

- Addresses a full-spectrum of internal and external events on a per plant-year basis
- Includes events that could affect multiple reactor modules to assess plant risk
- Uses ceramic fuel that will not melt when challenged by a full-spectrum of internal and external events
- Includes a “cliff edge” review to assure that the safety landscape is adequately addressed
- Assures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of the facility
- Provides successive compensatory means to prevent accidents or lessen the effects of damage if a malfunction or accident occurs
- Uses multiple, concentric, independent radionuclide barriers; breach of the helium pressure boundary does not result in failure of the fuel or the reactor building

MHTGR DBEs, DBAs, and BDBEs (aka EPBEs) on F-C Plot *(circa 1987)*



Other DBAs <10^{-8}

Staff Assessment of Next Generation Nuclear Plant (NGNP) Key Licensing Issues

Advisory Committee on Reactor Safeguards (ACRS)
Future Plant Designs Subcommittee Meeting

April 9, 2013

Donald Carlson, James Shea, Arlon Costa
Thomas Boyle, Jonathan DeGange

Office of New Reactors (NRO)
Division of Advanced Reactors and Rulemaking (DARR)

Assessment of NGNP Licensing Issues
Staff Presentations

ASSESSMENT OF NGNP LICENSING ISSUES – OVERVIEW

- Project Background, History, and Status
- Assessment Process and Staff Products

ASSESSMENT OF ISSUES IN 4 KEY AREAS

- Licensing Basis Event Selection
- Source Terms
- Functional Containment Performance
- Emergency Preparedness

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United States Nuclear Regulatory Commission
Protecting People and the Environment

NGNP Project History and Status

NGNP Project Mission, Energy Policy Act of 2005:

- Department of Energy (DOE) and Idaho National Laboratory (INL) will demonstrate by 2021 a prototype modular high temperature gas-cooled reactor (HTGR) for co-generating electricity and process heat
- NRC has licensing and regulatory authority for the prototype plant

Major NGNP Pre-Application Activities to Date

- Joint DOE-NRC Licensing Strategy Report to Congress, 2008:
  Option 2 risk-informed and performance based approach: Use deterministic engineering judgment and analysis, complemented by PRA insights, to establish NGNP licensing basis
- NRC assessment of DOE/INL white paper submittals, 2010-present

DOE decision in letter to Congress, October 2011:

- DOE will not proceed with NGNP detailed design activities at this time
- NGNP Project will continue to focus on high temperature reactor R&D, interactions with NRC to develop a licensing framework, and establishment of a public-private partnership

RESOURCES

- NRC has been using DOE reimbursable funds to assess NGNP licensing issues in 4 key areas

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Recent NGNP Interactions

NRC issued preliminary assessment reports to DOE, February 2012

- Assessment of Fuel Qualification and Mechanistic Source Terms (Rev. 0)
  - NGNP Fuel Qualification (FQ) White Paper
  - NGNP Mechanistic Source Terms (MST) White Paper
- Assessment of Risk-Informed and Performance-Based (RIPB) Approach (Rev. 0)
  - NGNP Defense-in-Depth Approach (DID) White Paper
  - NGNP Licensing Basis Event Selection (LBE) White Paper
  - NGNP Safety Classification of Structures, Systems, and Components (SSC) White Paper

NRC issued letter to DOE, February 2012

- Focus remaining NGNP interactions on issues in four key areas
  (1) Licensing Basis Event Selection (2) Source Terms
  (3) Functional Containment Performance (4) Emergency Preparedness

DOE/INL letter clarified approaches to key issues, July 6, 2012

- Public meetings and conference calls between NRC and DOE/INL, thru Nov 2012
- NRC staff review of supporting technical documents submitted by DOE/INL

DOE/INL provided information briefing to ACRS, January 17, 2013

Three Staff Products for NGNP

ISSUE SUMMARY REPORT

- Staff report: “Summary Feedback on Four Key Licensing Issues”

FQ-MST ASSESSMENT REPORT (REV. 1)

- Updated staff report: “Assessment of White Papers Submittals on Fuel Qualification (FQ) and Mechanistic Source Terms (MST).”

RIPB ASSESSMENT REPORT (REV. 1)

- Updated staff report: “Assessment of White Paper Submittals on Defense-in-Depth (DID), Licensing Basis Event (LBE) Selection, and Safety Classification of Structures, Systems, and Components (SSC).”

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Issue Summary Report

SUMMARY FEEDBACK ON FOUR KEY LICENSING ISSUES

i. Licensing Basis Event Selection
ii. Source Terms
iii. Functional Containment Performance
iv. Emergency Preparedness

- Issues highlighted in DOE-NRC NGNP Licensing Strategy Report to Congress (2008)
- Considered key issues in earlier NRC pre-application activities for proposed modular HTGRs, i.e., MHTGR (DOE/General Atomics,1986-1995) and Pebble Bed Modular Reactor (Exelon, PMBR Pty, 2001-05)
- All issues are considered in view of relevant prior staff positions, ACRS comments, and Commission direction (e.g., SECY-93-092, NUREG-1338, SECY-03-0047, SECY-05-0006, NUREG-1860, SECY-11-0152).
- The RIPB approach proposed for NGNP is similar to RIPB approaches that have been or may be considered for NUREG-1860, NUREG-2150, and NTTF Recommendation 1. A revised or new framework resulting from these other efforts may change the current NRC staff positions for NGNP.

Assessment of NGNP Licensing Issues

OVERVIEW

- After ACRS review, NRO will finalize the three staff products and publicly issue them to DOE
- Presentations today are based on the staff's Issue Summary Report

MAJOR CONCLUSIONS

- Staff views DOE/INL's proposed approaches to NGNP licensing issues as being generally reasonable, with caveats
  - Deterministic elements should be strengthened
  - Technical issues should be resolved through prototype testing under 10 CFR 50.43(e)(2)

QUALIFIERS

- Staff feedback is advisory; regulatory decisions will be based on NGNP license application and related Commission policy determinations
- Staff has assessed the proposed approaches solely as they apply to the modular HTGR design concept (next slides)

Early History of HTGRs

- **Dragon** - United Kingdom, 1966-75
  - Block type, 20 MWt, 750 ºC Outlet
- **AVR** - West Germany, 1967-88 →→
  - Pebble bed, 46 MWt (15 MWe), 950 ºC
- **Peach Bottom 1** - United States, 1967-74
  - Block type, 115 MWt (40 MWe), 725 ºC Outlet

- **Fort St. Vrain** - United States, 1976-89
  - Block type, 840 MWt (330 MWe), 785 ºC Outlet
- **THTR** - West Germany, 1985-89
  - Pebble bed, 750 MWt (300 MWe), 750 ºC Outlet

Image courtesy of Juelich Research Centre

HTGR Design Evolution (U.S. & Germany)
Post-TMI Shift to Modular HTGR Safety Concept

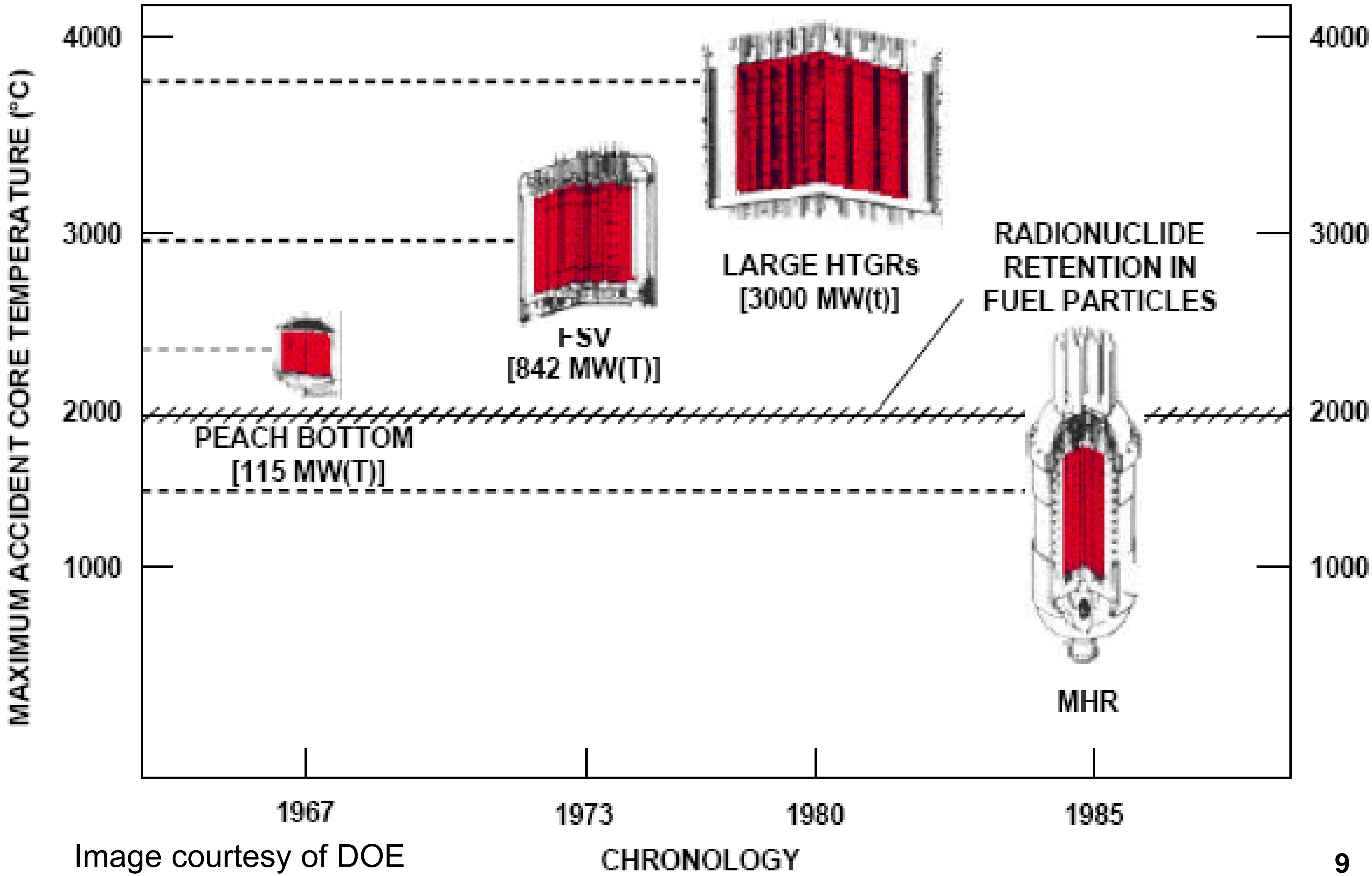
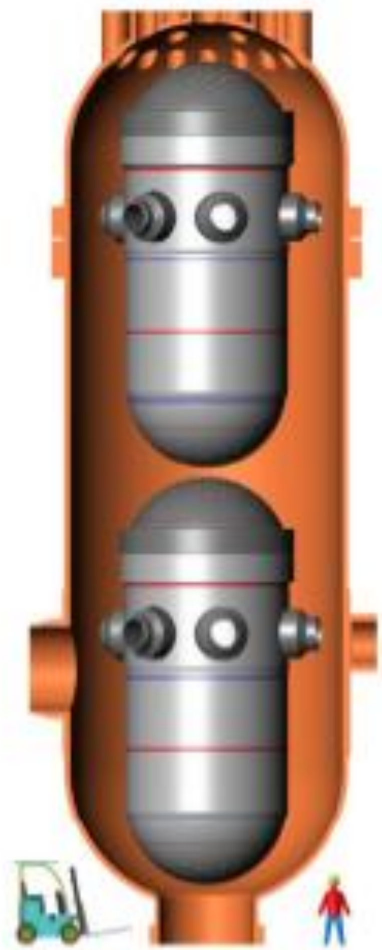


Image courtesy of DOE

Modular HTGR Design Concept

Size Comparison

- Per unit power output, modular HTGRs are much larger than LWRs
- Relative to LWRs, modular HTGRs have
  - Much lower core power density
  - Much lower fuel volume fraction in active core
    - LWRs ~30%
    - HTGRs ~0.5%
  - Much greater thermal inertia



600 MWt HTGR RPV
vs
PWR RPV

Image courtesy of DOE

LICENSING BASIS EVENT SELECTION

Licensing Basis Event Selection

SUMMARY OF DOE/INL PROPOSAL

- DOE/INL proposes a process for selecting and evaluating NGNP licensing basis event sequences (LBEs) that seeks to blend the strengths of probabilistic and deterministic methods
- The process would yield LBEs categorized as Anticipated Events, Design Basis Events, Design Basis Accidents, and Beyond Design Basis Events
- Offsite dose consequences of LBEs would be evaluated and assessed against Top Level Regulatory Criteria (TLRC) and EPA Protective Action Guidelines (PAGs) placed on a Frequency-Consequence (F-C) curve
- The LBE process would incorporate a risk-informed approach to safety classification of structures, systems, and components (SSCs)

Licensing Basis Event Selection

DOE/INL's Proposed Frequency-Consequence Curve

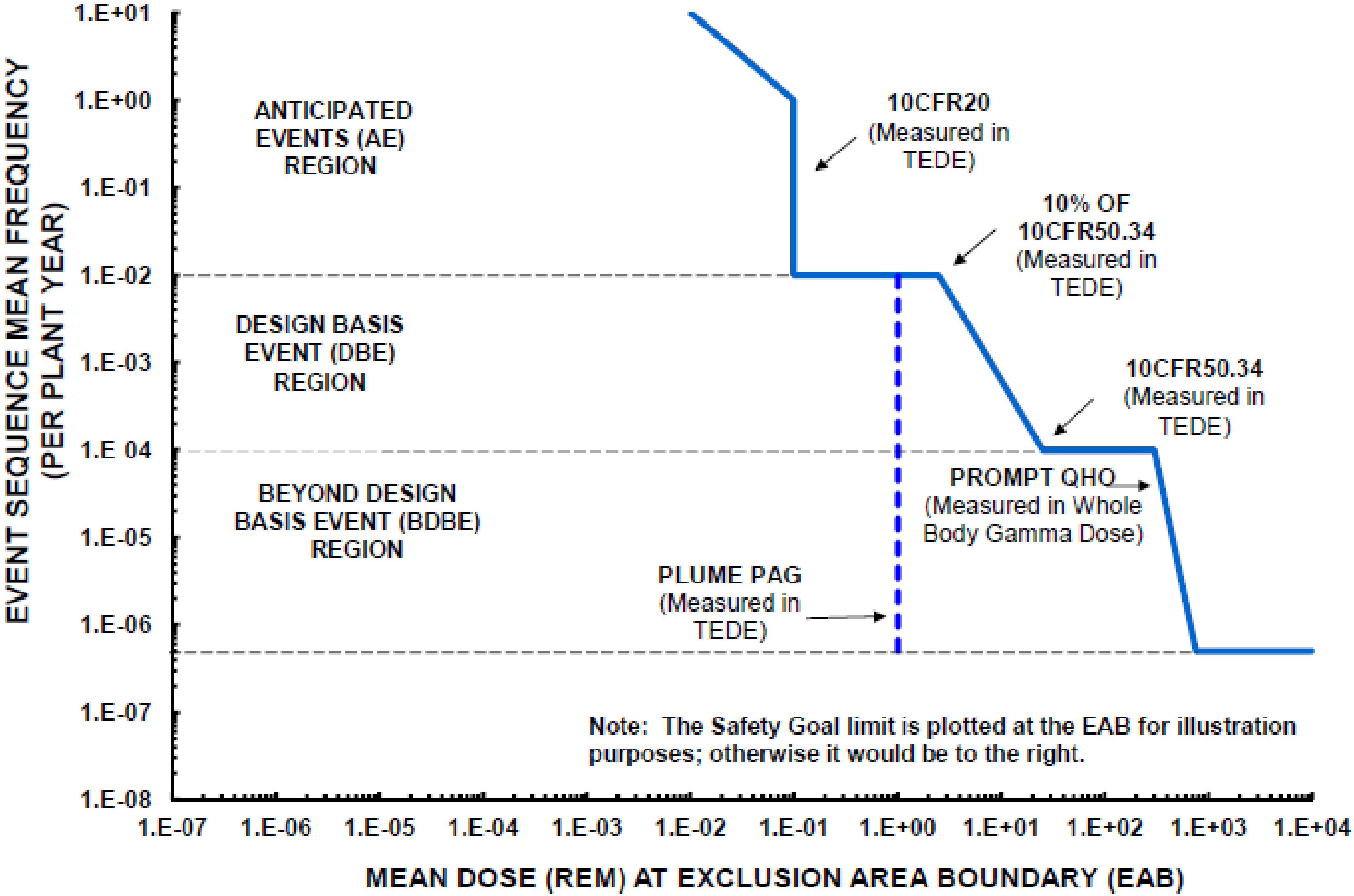


Image courtesy of DOE

Licensing Basis Event Selection

SUMMARY OF STAFF FEEDBACK ON LBE SELECTION

- Proposed LBE selection approach is generally reasonable but overly risk-based in some respects. Deterministic elements should be strengthened
- Future Commission direction may be appropriate for issues such as:
  - Frequency cutoffs for Design Basis and Beyond Design Basis Events
  - "Per-plant-year" method for addressing risk at multi-reactor module plant sites
  - Process and criteria used for selection of DBAs to demonstrate regulatory compliance
  - Consideration of alternate TLRC and F-C curves (e.g., NUREG-1860) in the contexts of
    - Future licensing of NGNP or other modular HTGRs
    - Developing a Technology Neutral Framework, etc.

Licensing Basis Event Selection

Issue 1: DOE/INL requests NRC agreement on key terminology and naming conventions for its proposed event categories.

NRC Staff Feedback

- Proposed event category names and descriptions are reasonable
- Full set of approved LBEs may have to include more deterministic events
  - Postulated DBAs and BDBE-derived DBAs in addition to DBAs derived from DBEs
  - AEs evaluated against specified acceptable fuel/core design limits (SAFDLs) for HTGR
- Final selection of DBAs may need to include postulated deterministic event sequences

Licensing Basis Event Selection

Issue 2: DOE/INL requests that NRC endorse the proposed process and categorizations for SSC classification.

NRC Staff Feedback

- Approach blends the strengths of probabilistic and deterministic methods in accordance with the NRC's policy statement on PRA
- Applies a risk-informed approach while addressing traditional deterministic definition of safety-related SSCs in 10 CFR 50.2
- Special treatments for the safety-related and non-safety-related with special treatment (NSRST) categories of SSC classification commensurate with ensuring that SSCs can perform required safety functions for LBEs, provide DID
- Processes and categorizations for SSC safety classification are reasonable

Licensing Basis Event Selection

Issue 3: DOE/INL requests NRC agreement with its proposed placement of TLRC on an F-C curve.

NRC Staff Feedback

- The selected TLRC and their placement on an F-C curve are reasonable
- DOE/INL should pursue an appropriate regulatory limit to ensure the required level of integrity of the fuel barrier
- Deterministic elements of the proposed approach should be strengthened.
- Future Commission direction may be appropriate for determination of dose acceptance criteria for various event categories

Licensing Basis Event Selection

Issue 4: DOE/INL requests that NRC establish frequency ranges based on mean event sequence frequency.

NRC Staff Feedback

- The approach for categorizing each event sequence based on mean frequency is reasonable
  - Uncertainties would be considered in deriving both mean frequency and mean consequence of event sequences
  - Upper (95%) and lower (5%) bounds of the event frequency uncertainty distribution will be compared against the frequency boundaries of the LBE categories
  - If the upper or lower bounds of confidence intervals straddle frequency boundaries between LBE categories, the consequences of the event sequence will be compared against the criteria for each LBE category

Licensing Basis Event Selection

Issue 5: DOE/INL requests that NRC endorse the "per-plant-year" method for addressing risk at multi-reactor module plant sites.

NRC Staff Feedback

- Proposed "per plant-year" method for addressing risk at multi-module plants is reasonable
- The staff believes that an integrated risk approach is more conservative and comprehensive than the treatment of modules on an individual basis
- Proposed method would appropriately address event sequences that involve source terms from one reactor module or multiple reactor modules
- Future Commission direction may be appropriate for this topic

Licensing Basis Event Selection

Issue 6: DOE/INL requests that NRC agree on the frequency cutoffs for the DBE and BDBE regions.

NRC Staff Feedback

- Top design objective is to meet the EPA PAGs at the site boundary for all event sequences more frequent than 5E-7 per plant year
- DOE/INL provide justification for frequency cutoffs in their LBE white paper
- Uncertainties would be considered in deriving both the mean frequency and mean consequence of event sequence
- Staff believes frequency cutoffs are reasonable for modular HTGRs as long as the PRA used in the LBE selection process:
  - assesses multiple failures from common-cause events
  - account for both operating and shutdown modes, internal and external plant hazards
- Future Commission direction may be appropriate for deciding frequency cutoffs for modular HTGR licensing

Licensing Basis Event Selection

Issue 7: DOE/INL requests that NRC endorse the overall process for performing assessments against TLRC such as issues with uncertainties and PRA, calculational methodologies employed, and adequate incorporation of deterministic elements.

NRC Staff Feedback

- DOE/INL's proposed approach to using engineering judgment to address uncertainties is a reasonable approach for assessing LBEs in a risk-informed manner
  - LBEs with frequency uncertainty distributions that straddle two event category regions at the 95% confidence level would be analyzed using the dose acceptance criteria of each region
  - Calculational methodologies to be employed assess full event sequences using best-estimate models with mean or conservative analysis (95% confidence)
- For AE and BDBE compliance with the TLRC, the proposed approach of realistic source term calculations needs further consideration and would involve new regulatory interpretations for potential future consideration by the Commission
- Certain elements of the proposed approaches are overly risk-based. Deterministic elements should be strengthened

Licensing Basis Event Selection

NRC Staff Feedback cont'd

The process is described by DOE/INL as:

- Technology neutral
- Comprehensive; considers full plant response to a wide spectrum of events
- Quantitative; so safety margins can be assessed

Proposed approaches are generally consistent with relevant past staff positions and Commission guidance such as:

- Advanced Reactor Policy Statement
- NUREG-1338 MHTGR PSER
- NUREG-1860 Feasibility Study for Performance-Based Reg. Structure
- SECY-93-092, SECY-95-299, SECY-98-0300
- SECY-03-0047, SECY-04-0157, SECY-05-006

MECHANISTIC SOURCE TERM

Mechanistic Source Term

BACKGROUND

- Commission approved the use of event-specific mechanistic source term (MST) as proposed in SECY-93-092 and SECY-03-0047

> *A mechanistic source term is the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.* [SECY-93-092]

> *[T]he use of scenario-specific source terms [is allowable], provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties.* [SECY-03-0047]

Mechanistic Source Term

SUMMARY OF STAFF FEEDBACK

- The NRC staff's overall assessment is that the proposed approaches to mechanistic source terms are generally reasonable, with some potentially significant caveats

Mechanistic Source Term

Issue 1: 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.

NRC Staff Feedback

- Consistent with SRMs to SECY-93-092 and SECY-03-0047
- The NRC staff concludes that DOE/INL's proposed definition of NGNP mechanistic source terms aligns with the current staff position on the treatment of advanced reactor mechanistic source terms and is thus reasonable for use in DOE/INL's proposed approach to determining licensing parameters for modular HTGRs

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Mechanistic Source Term

Issue 2: Agree that NGNP source terms are event specific and determined mechanistically using models of radionuclide generation and for transport that account fuel and reactor design characteristics, passive features, and the radionuclide release barriers.

NRC Staff Feedback

- Consistent with SRMs to SECY-93-092 and SECY-03-0047
- DOE/INL's proposed approaches to developing event-specific mechanistic source terms are reasonable

Mechanistic Source Term

Issue 3: 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.

NRC Staff Feedback

- Ongoing and planned testing and research activities for NGNP fuel qualification and mechanistic source terms development are generally reasonable
- Staff expects more information on release and transport phenomena through event-specific pathways to be developed as DOE/INL's activities in these areas proceed
- Data from NGNP prototype tests would be needed to verify and supplement the technical basis for NGNP fission product transport modeling and validation

Mechanistic Source Term

DOE/INL proposes to include in their fission product transport models:

- Transport of radionuclides from their point of origin through the fuel to the circulating helium
- Circulating activity in the helium pressure boundary (HPB)
- Distribution of condensable radionuclides in the HPB (plateout and dust)
- Radionuclide release from HPB and distribution in the reactor building (i.e., circulating activity, lift-off, wash-off; heat-up)
- Radionuclide release from the reactor building to the environment (source term)
- In addition to providing source terms, these calculations provide radionuclide inventories throughout the plant.

Mechanistic Source Term

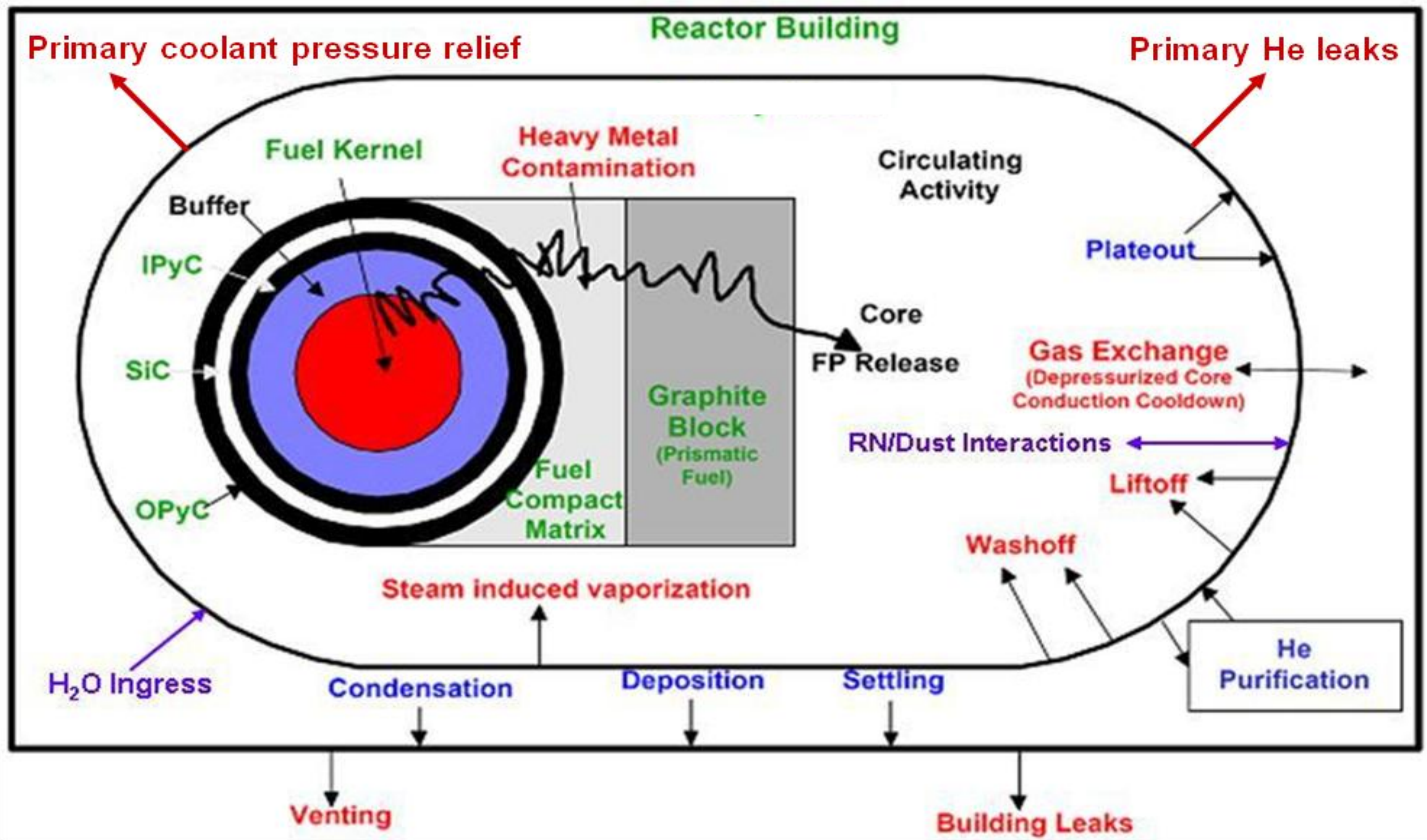

Image courtesy of DOE

Mechanistic Source Term

ADDITIONAL NRC STAFF FEEDBACK

The NRC staff's overall assessment is that the proposed approaches to mechanistic source terms are generally reasonable, with some potentially significant caveats.

- Staff's preliminary view is that some fuel qualification elements should to be supplemented to support the MST and the NGNP safety case
- The NRC staff believes satisfactory completion of a post-irradiation fuel inspection and testing program including fuel from an NGNP prototype is necessary to verify and supplement the technical basis for NGNP MST code validation
- The draft ASME/ANS PRA standard states that it is required that all PRA elements (including the mechanistic source term element) have a peer review. The staff views such peer review as having particular importance for the implementation of risk-informed approaches to NGNP licensing

Mechanistic Source Term

ADDITIONAL NRC STAFF FEEDBACK

- LBEs for siting should include postulated bounding events that adequately challenge all available barriers in the assessment of event-specific mechanistic source terms. Postulated events could include:
  - Bounding events with air ingress
  - Bounding events with water ingress
- Safety terrain studies related to BDBEs of low probability should be evaluated to inform the selection of LBEs used in establishing the EPZ and EP requirements
- Staff believes that DOE/INL's Research Plan for Moisture and Air Ingress (PLN-4086, April 2012) presents a reasonable approach for providing data needed for developing and validating models for predicting the effects of air and moisture ingress on NGNP TRISO fuel performance and fission product transport
- In SECY 05-0006, the staff recommends that source terms for compliance should be 95% confidence level values based on best-estimate calculations

FUNCTIONAL CONTAINMENT PERFORMANCE

Functional Containment Performance

OVERVIEW OF DOE/INL PROPOSAL

- Proposed definition of 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*
  - *NRC regulatory requirements and plant design goals for release of radionuclides are met at the Exclusion Area Boundary*

- DOE/INL requests NRC feedback on three elements of its approach to NGNP functional containment:
  - AGR Fuel Program activities
  - Options for containment functional performance standards
  - Event selection for plant siting and functional containment design decisions

Functional Containment Performance

Issue 1: AGR Fuel Program Activities

Confirm that plans being implemented in AGR Fuel Program are generally acceptable and provide reasonable assurance that TRISO fuel can retain fission products in predictable manner. Identify any additional information or testing needs.

NRC Staff Feedback - Overview

- Scope of AGR activities is generally reasonable in context of pre-prototype testing
- Early AGR irradiation and safety testing results show promise for demonstrating much of desired TRISO fuel retention capability
- Additional data are needed from fuel and core testing in NGNP Prototype to provide reasonable assurance of targeted fission product retention in fuel
  - Test data on fuel irradiated in HTGR for effects of plutonium fission products (Pd, Ag) on TRISO particle coatings
  - Testing in prototype to confirm NGNP core operating conditions and ability to detect potential core “hot spot” operating anomalies
- 10 CFR 50.43(e)(2) allows NRC to impose additional requirements on prototype plant during testing period

Functional Containment Performance

Issue 1: AGR Fuel Program Activities (cont.)

Additional Staff Feedback

- Adequately define fuel service conditions and performance requirements
  - Normal operations
    - Pu burnup for potential effects of Pu fission products (Pd, Ag) on TRISO fuel particle coatings
    - Potential effects of irradiation parameter path dependence
    - NGNP core operating condition uncertainties and anomalies (hot spots)
  - Accidents
    - Design information is needed to confirm DOE/INL's assumed lack of specific fuel testing requirements for reactivity excursion events
    - DOE/INL's Research Plan for Moisture and Air Ingress (April 2012) should be implemented to address data needs for fuel performance and fission product transport in bounding events
- Supplement AGR data with data from fuel irradiated in NGNP prototype
  - Real-time versus accelerated testing
  - Prototypic plutonium burnup

Functional Containment Performance

Issue 1: AGR Fuel Program Activities (cont.)

Additional Staff Feedback

- Evaluation of irradiation test temperature uncertainties
  - Additional information provided in two INL submittals
    - "AGR-1 Thermocouple Analysis," May 2012
    - "Uncertainty Quantification of Calculated Temperatures for AGR-1 Experiment," June 2012
  - Important to understand how AGR irradiation temperature uncertainties are quantified and affected by increasing thermocouple failures
- Assessment of applicability of delayed fuel heatup testing
  - DOE/INL report (TEV-1543, June 2012) analyzes potential changes in fuel composition during the interim between irradiation and heatup testing. Results support application of data from delayed fuel heatup tests to the modeling of fuel performance and fission product transport in NGNP accidents.

Functional Containment Performance

Issue 2: Options for Containment Functional Performance Standards

NRC Staff Feedback

- Proposed approach presents a reasonable option for establishing modular HTGR functional containment performance standards (per SRM to SECY-03-0047)
  - Radionuclide containment function: reduce releases to the environs
  - Other "containment" functions as discussed in SECY-05-0006
    - Protect risk-significant SSCs from internal and external events
    - Physically support risk-significant SSCs
    - Protect onsite workers from radiation
    - Remove heat to keep risk-significant SSCs within design and safety limits
    - Provide physical protection (i.e., security) for risk-significant SSCs
    - Reduce radionuclide releases to environs (including limiting core damage)
    - Limit air ingress after helium depressurization accidents
- Future Commission policy direction may be appropriate for determining specific criteria applied to modular HTGR functional containment

Functional Containment Performance

Issue 3: Event selection for plant siting and functional containment design decisions

NRC Staff Feedback

- Core melt accident assumed for LWRs may not be applicable to modular HTGRs
- Proposed approach to event selection for siting source terms is generally reasonable when supplemented with insights from "safety terrain" studies
  - *Applicant should submit for NRC consideration a risk-informed selection of siting events, building on the types of bounding events considered by staff in NUREG-1338 for MHTGR*
  - *To assure there are no "cliff-edge effects" [credible events with high dose consequences] and to understand ultimate safety capability, bounding event selection should be further informed by exploratory studies of postulated extreme events, including bounding events with air oxidation of graphite per the SRM to SECY-93-092. Such exploratory events should be physically plausible, may have estimated frequencies below the BDBE region (< 5E-7), and will consider inherent behavior of the modular HTGR design*
- Future Commission direction may be appropriate for the selection of siting source term events for functional containment criteria

EMERGENCY PREPAREDNESS

Emergency Preparedness

BACKGROUND

- In October 2010, DOE/INL submitted a white paper on "Determining the Appropriate Emergency Planning Zone Size and Emergency Planning Attributes for an HTGR"
  - NRC staff did not formally review this white paper submittal and provided no formal feedback to DOE/INL on its contents
  - DOE/INL participated in NRC public meetings in 2011 on emergency preparedness framework issues for small modular reactors (SMRs)
- In October 2011, NRC staff issued SECY-11-0152, Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors
  - Described a general approach to scalable EPZs
- 10 CFR 50.47(c)(2) allows Emergency Planning Zone (EPZ) size for gas-cooled reactors to be considered on a case-by-case basis

Emergency Preparedness

SUMMARY OF STAFF FEEDBACK

- DOE/INL's proposed approaches are generally reasonable and are responsive to the Commission's Policy Statement on Advanced Reactors
- Future Commission direction may be appropriate for this area
- The staff is open to considering alternative EP requirements and frameworks for advanced reactors and SMR facilities
- The staff does not plan to propose additional new EP policies or to revise the existing guidance for addressing EP requirements at this time

Emergency Preparedness

Issue 1: DOE/INL requests that NRC propose a new policy or revised regulations on EPZ sizing. NGNP goal is to justify EPZ at 400-meter Exclusion Area Boundary.

NRC Staff Feedback:

Consistent with SECY-11-0152

- Describes dose-distance scalable EPZ approach
- Staff will more fully address EP issues in the context of site-specific pre-application reviews
- Staff would be open to considering future proposals from industry or established pre-applicants on such topics as:
  - PRA-informed approach that includes dose assessment versus distance
  - Risk-informed criteria for determining the point at which the probability of exceeding the PAG values is acceptably low

Emergency Preparedness

Issue 2: DOE/INL requests that NRC establish specific guidance on graded approaches to applying EP requirements in relation to the PAGs.

NRC Staff Feedback:

- NRC expects specific proposals from NGNP pre-applicant to be supported by details of the NGNP design, site, and co-located user facilities
- Proposed EP approaches for NGNP should include consideration of how potential EP basis events may be influenced by co-location and external events impacting the site
- NGNP EP approach addressing PAGs must be developed by the site applicant
- Graded EP may be different for NGNP prototype plant versus subsequent standard plants

Emergency Preparedness

Issue 3: DOE/INL requests that NRC propose guidance on how issues related to modularity of the designs and the co-location of multi-module plants near industrial facilities should be considered in EP.

NRC Staff Feedback:

- Co-location considerations for current LWRs are largely applicable
  - Regulatory guidance already incorporated in existing EP plans
- Co-generation implying co-located utilization of nuclear heat sources
  - Different regulatory nexus
  - Safety strategy must consider challenges and issues arising from the modular HTGR being coupled to the industrial facility
- Expect staff considerations of new regulations, hazards assessments, accident evaluations, and security issues

THANK YOU