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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

FUTURE PLANT DESIGN SUBCOMMITTEE

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NGNP RESEARCH AND LICENSING ISSUES

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THURSDAY, JANUARY 17, 2013

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

DENNIS C. BLEY, Chairman

J. SAM ARMIJO, Member

CHARLES H. BROWN, JR. Member

MICHAEL L. CORRADINI, Member

HAROLD B. RAY, Member

JOY REMPE, Member

WILLIAM J. SHACK, Member

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ACRS CONSULTANTS PRESENT:

THOMAS A. KRESS

NRC STAFF PRESENT:

MAITRI BANERJEE, Designated Federal Official

DON CARLSON, NRO

JIM SHEA, NRO

ALSO PRESENT:

DAVID ALBERSTEIN, INL

DAVID HANSON, INL

MARK HOLBROOK, INL

JIM KINSEY, INL

DAVID PETTI, INL

FARSHID SHAHROKHI, AREVA

FRED SILADY, INL

CARL SINK, DOE

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(8:31 a.m.)

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

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

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

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

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

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1 related to their work.

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

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

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

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

24 I also want to mention, we have a really
25 tight schedule today, and a lot of material to go

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1 over. We have to stop earlier than normal because of
2 a separate meeting that was scheduled for later this
3 afternoon. So we have to finish by 3:00.

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

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

13 MR. CARLSON: Yes.

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

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

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

24 Our plan is to finalize some feedback to
25 DOE and INL on some of these issues, and present that

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1 to the subcommittee in a few months. So the purpose
2 of today's briefing, it's an information briefing, and
3 it's between the ACRS members and DOE/INL.

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

8 MEMBER CORRADINI: Can I ask a question
9 then?

10 CHAIR BLEY: Certainly.

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

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

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

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

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

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

13 MEMBER CORRADINI: Okay.

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

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

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

23 MEMBER CORRADINI: Can I ask one other
24 thing?

25 MEMBER ARMIJO: When would we hear about

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1 your feedback report? When would we see it?

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

5 MS. BANERJEE: April 9th.

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

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

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

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

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

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1 concurrence process at that time.

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

4 MEMBER CORRADINI: They will through
5 concurrence now.

6 MR. CARLSON: They will.

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

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

14 MR. CARLSON: Exactly.

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

18 CHAIR BLEY: Okay.

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

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

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

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1 (Laughter)

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

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

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

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

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

22 The Nuclear Energy Advisory Committee in
23 2010 and 2011 reviewed the status of the NGNP project.
24 As part of that, they looked at the status of our
25 regulatory development and, in their final report to

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1 the Secretary, recommended that we continue our
2 interactions with the Nuclear Regulatory Commission to
3 develop the licensing framework.

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

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

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

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

25 MR. CARLSON: No. There's no official

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1 position by DOE between prismatic or pebble.

2 MEMBER REMPE: Okay. And also, what's
3 going on? I saw the alliance wanted to do something
4 now in Georgia they're talking about maybe building.
5 Is that anything significant, or it's still just very
6 --

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

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

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

17 MR. CARLSON: Sure.

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

23 MR. CARLSON: Actually, no. That was a
24 report specific to modular reactor safety base. It
25 was submitted for information only. And from the

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1 staff, it was requested that we not have any public
2 meetings on that yet.

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

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

8 MEMBER CORRADINI: Oh, the modular
9 aspects.

10 MR. CARLSON: The modular aspects --

11 MEMBER ARMIJO: Excuse me, multiple.

12 DR. SINK: Multiple modules that are --

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

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

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

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1 modular HTGR.

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

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

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

11 MEMBER CORRADINI: Oh, fine. Thank you.

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

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

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

18 MEMBER ARMIJO: Yes. We can look at
19 that.

20 MEMBER CORRADINI: Thank you.

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

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

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

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

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

25 I just want to restate that DOE is

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

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

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

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

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

22 MEMBER CORRADINI: Since you guys have
23 just awarded the SMR, I don't remember the right
24 title for it, but to the B and W, mPower design, is
25 it expected that mPower is going to use some of these

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1 analyses and discussions as a basis for their
2 discussions with the staff?

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

6 MEMBER CORRADINI: Okay.

7 DR. SINK: Any other questions?

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

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

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

21 MEMBER CORRADINI: What do you define as
22 strong?

23 DR. SINK: Well --

24 MEMBER CORRADINI: So I'm just trying to
25 understand. If there's a gap, I want to understand

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1 what your expectation is versus what we will hear in
2 April.

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

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

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

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

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

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

18 MEMBER CORRADINI: Okay, thank you.

19 MEMBER REMPE: Here, this letter.

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

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

25 DR. KRESS: Which of these five issues do

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1 the design basis accidents fall under?

2 MEMBER CORRADINI: The LBES.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

23 The particles are very small. You've
24 seen them. I forgot to bring my little show and tell
25 hand out to pass around. They're compacted into a

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1 matrix in graphite, within the fuel element, either
2 form, either as a pebble or block. So that composes
3 a fuel element, and there's pictures to come.

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

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

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

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

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

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

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

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

13 This is the reactor building below grade.
14 It includes the three vessels. Again, it encloses,
15 so these are nested barriers. It completely encloses
16 the three vessels, the reactor vessel, the cross
17 vessel, and the steam generator vessel.

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

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

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

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

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

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

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

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

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

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

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

23 MR. SILADY: Yes, that's correct. Next
24 page. So that means some things to the design and to
25 the R and D. It means that this is almost a repeat,

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1 same three bullets parallel to those that we talked
2 about before.

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

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

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

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

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

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

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

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

22 Next page. I need to speed it up just a
23 little bit here. What I'm going to do now is take
24 those three functions at the bottom, the passive heat
25 removal, and control heat generation and chemical

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1 attack, and use that as a mini outline for the next
2 three or four slides.

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

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

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

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

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

22 MEMBER ARMIJO: Okay.

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

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

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

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

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

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

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

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

10 MR. SILADY: It was also --

11 (Off microphone comments)

12 MR. SILADY: 6.3.

13 MEMBER CORRADINI: 6.3, thank you.

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

16 MEMBER CORRADINI: Okay, thank you.

17 MR. SILADY: Next page.

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

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

25 Next page. These are the transients for

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

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

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

16 MEMBER CORRADINI: So, this is from the
17 '89 analysis?

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

20 MEMBER CORRADINI: I was just guessing.

21 MR. SILADY: Yes, that's correct.

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

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

4 MEMBER CORRADINI: Okay.

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

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

10 MEMBER CORRADINI: Thank you.

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

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

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

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1 1600. And that's why the average that's shown below
2 in the previous slide was lower.

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

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

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

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

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

23 And you can see up to 500 hours now, at
24 1600 constant. And we were never at 1600 in the
25 transient, but maybe 50 hours. And yes, if you go to

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1 1700, maybe you're starting to get a particle failure
2 there, 1800 you did, 2100 you did, hundreds of degree
3 margin to where we're designing it. The rule of
4 thumb is 1600 C. Next page.

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

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

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

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

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

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

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

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

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

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

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1 flow area and friction losses.

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

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

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

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

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

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

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

11 MEMBER REMPE: Before you go on --

12 MR. SILADY: Yes?

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

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

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

25 PARTICIPANT: No.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5 MEMBER ARMIJO: Okay. So you would form
6 hydrogen.

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

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

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

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

25 MEMBER SHACK: Here it says within and

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

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

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

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

15 CHAIR BLEY: Thanks, Fred.

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

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

25 And then Fred will come back in and talk

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1 some more, and provide some of the detail behind the
2 presentation that he just gave.

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

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

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

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

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

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

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

16 MEMBER ARMIJO: I'm sorry.

17 MR. HOLBROOK: Let's go back.

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

21 MR. HOLBROOK: I think --

22 MEMBER ARMIJO: The request to the --

23 MR. HOLBROOK: Yes, this --

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

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

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

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

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

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

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

25 At a very high level, the framework needs

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1 to answer these kinds of questions. What must be
2 met? What criteria must be met? When must that
3 criteria be met? How we're going to meet them and
4 how well do we have to meet them?

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

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

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

16 MR. HOLBROOK: Yes.

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

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

25 We're looking for things that were

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

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

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

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

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

25 As we've already mentioned, we're taking

1 into consideration the PAGES, the one rem TEDE limit
2 evaluated at the EPZ. That'll be our design limit
3 for the plant, our design goal, I should say.

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

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

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

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

15 MEMBER ARMIJO: Right out of the
16 regulation?

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

22 MEMBER ARMIJO: Got it.

23 MEMBER CORRADINI: So --

24 MR. HOLBROOK: Yes, sir?

25 MEMBER CORRADINI: Can we go back?

1 MR. HOLBROOK: Yes.

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

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

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

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

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

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

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

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

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

25 MEMBER CORRADINI: I understand. I just

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1 wanted to be sure.

2 (Crosstalk)

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

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

8 MEMBER CORRADINI: And you are?

9 MR. SHEA: Jim Shea of the staff.

10 (Off microphone comments)

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

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

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

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

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

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

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

23 We're expressing these frequencies on a
24 per-plant-year basis. Obviously this is what's most
25 important to the public. They want to know what's

1 the impact on them. They don't care if it's this
2 plant or that plant.

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

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

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

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

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

25 And finally, on this slide here in this

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

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

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

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

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

25 As the initial design is developed, you

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

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

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

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

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

22 DR. KRESS: Do you have your PRA already?

23 MR. HOLBROOK: Say again?

24 DR. KRESS: Do you have a PRA yet?

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

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1 point in --

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

4 MR. HOLBROOK: Yes.

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

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

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

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

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

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

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

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

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

6 MR. HOLBROOK: It's pretty similar.

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

12 MEMBER CORRADINI: Let me just make sure
13 I say it right. Maybe I'm saying it wrong. But in
14 '89 you used 10 CFR 50. You did DVAs. You didn't
15 have to deal with severe accidents in this
16 standpoint.

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

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

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

4 MEMBER CORRADINI: Okay, all right.

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

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

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

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

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

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1 their own accident rule set.

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

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

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

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

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

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

25 MR. SILADY: Yes, if you could shift it

1 a little bit.

2 MEMBER ARMIJO: I'd like to understand
3 that.

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

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

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

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

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

21 CHAIR BLEY: Okay, I thought you said --

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

24 CHAIR BLEY: Ah.

25 MR. SILADY: And the reason we added it

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

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

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

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

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

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

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

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1 handled.

2 DR. KRESS: I see.

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

5 MR. SILADY: Okay.

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

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

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

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

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

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

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

25 MR. SILADY: Yes, seismic would be a good

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1 one, loss of offsite power would be another one that
2 affects all four.

3 DR. KRESS: I see.

4 MR. SILADY: Or there could be subtle
5 differences. And they're all maintained by the same
6 --

7 MEMBER CORRADINI: I get it.

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

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

18 MR. SILADY: We don't have understanding.

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

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

24 And then we plot what that frequency is
25 for that one, two, three, or four on there. And

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1 sometimes it's multiplication. A lot of times it's
2 not. So we leave that to be figured out in terms of
3 the event frequency.

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

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

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

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

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

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

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

24 So you don't do the multiplication. And
25 that's what I'm fighting against here, kind of coming

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1 back a little bit. So I see your point. But I want
2 to be more general.

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

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

9 DR. KRESS: A factor of ten.

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

15 And in the MHTGR days, we designed 40
16 year lifetime. Now, these days, the NGNP is 60 year.
17 So rather than seeing this thing move, and having a
18 weird number that wasn't round, we decided to go to
19 10 to the minus 2 for the anticipated events.

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

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1 rounding, if you will.

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

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

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

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

20 DR. KRESS: Why could you not?

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

25 DR. KRESS: You could have a conservative

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1 line.

2 MR. SILADY: Sure.

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

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

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

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

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

23 We have to know what that is so that we
24 know that it has the right capability, and the right
25 reliability, and so on. So events that are close to

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1 the line are important. But events with zero dose
2 are important as well.

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

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

9 And we all know that average background
10 natural is 300 millirem. And you have another 300
11 from man-made causes. So it's a very tight limit
12 that is put on nuclear power plants, incrementally
13 over the background, that people get, which would be
14 600 millirems, average.

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

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

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

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

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

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

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

21 MR. SILADY: Yes?

22 MEMBER REMPE: And I decide to put four
23 modules in. And later the utility, or the
24 owner/operator, I guess is what Harold calls it,
25 decides to build four more modules. Does that mean

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1 the requirement on my first four modules needs to be
2 a bit tighter, because suddenly you've got eight
3 modules that have got to meet this.

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

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

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

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

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

25 But let's say some other technology

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

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

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

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

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

19 MR. SILADY: That's correct.

20 CHAIR BLEY: Okay, that's your thinking.

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

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1 us?

2 CHAIR BLEY: Exactly right.

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

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

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

15 MR. SILADY: An isotherm, a risk?

16 DR. KRESS: No risk, in reverse.

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

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

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

25 When we got the feedback from the NRC at

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1 that time, we're proposing this. If somebody thinks
2 it should be 5 percent, or 25 percent, fine, if we
3 think it's got a sound basis that'll last for the
4 duration of the project.

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

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

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

22 It's prompt depth. So the line goes flat
23 at 5 times 10^{-7} . We could find no other
24 hint in the regulations where the de minimis should
25 be. It has to be there somewhere.

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

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

10 And again, we're proposing. And if
11 somebody wants this to go to 2 times 10 to the minus
12 7, or 10 to the minus 7, fine. If the axis goes 10
13 to the minus 8, we do our PRAs down to 10 to the
14 minus 8, and try to be as complete as possible.

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

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

24 DR. KRESS: When I first saw the curve I
25 wondered why the slope of that slanting part was

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1 different than the other slopes.

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

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

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

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

22 MEMBER CORRADINI: Sir, I'm sorry.

23 MR. SILADY: Sure.

24 MEMBER CORRADINI: If you could just
25 repeat, the dashed line you're going to meet how

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1 again, I'm sorry?

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

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

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

8 MEMBER CORRADINI: Yes.

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

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

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

25 And I take the design basis accidents,

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1 the design basis events, the ones that have the
2 entire plant responding, that are from that plot, and
3 I take the three dominant ones. They happen to be 6,
4 10, and 11.

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

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

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

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

23 The next one is that vessel system relief
24 line having a breach. This is about a 13 square inch
25 line. And so now you get, in addition, an immediate

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1 and definite loss of forced cooling in this
2 particular accident.

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

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

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

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

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

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

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

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

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

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

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

23 We back up, and in just a couple of
24 slides I'm going to show you the process, or how we
25 determine what's safety related. But given that we

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1 know what's safety related, we then go back to DBE-6,
2 10, and 11, which are up in the DBE space as black.
3 And we say now we're only going to have the safety
4 related SSCs respond. Those are the green points.

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

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

10 MEMBER CORRADINI: Oh, okay.

11 MEMBER REMPE: So the moisture --

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

14 MEMBER REMPE: Were not available for
15 DBA.

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

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

22 MR. SILADY: The frequency stays the
23 same, depending if the design basis event happened to
24 only have safety related equipment. Or it goes down
25 if it had other things like the moisture monitors, or

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1 the shutdown cooling system coming on.

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

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

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

12 MR. SILADY: Okay.

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

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

17 MEMBER CORRADINI: That's eight.

18 MR. SILADY: No, it is --

19 MEMBER REMPE: Go up a bit.

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

21 MEMBER CORRADINI: No problem, all right.

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

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

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

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

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

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

18 MEMBER CORRADINI: Thank you.

19 MR. SILADY: Okay.

20 MEMBER CORRADINI: I got it.

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

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

24 (Off microphone comments)

25 MR. SILADY: It's down here somewhere.

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1 MEMBER CORRADINI: Okay, fine.

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

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

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

11 MS. BANERJEE: I know, on the --

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

15 MEMBER ARMIJO: What is DC-1?

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

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

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1 going to be an EPBE.

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

6 MEMBER CORRADINI: So since we're
7 noodling --

8 MR. SILADY: Yes, that's fine.

9 MEMBER CORRADINI: What's WC-1?

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

14 (Laughter)

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

20 MEMBER CORRADINI: And W means wet, means
21 what?

22 MR. SILADY: Water, water comes in.

23 MEMBER CORRADINI: Oh, it's flooded.

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

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1 MEMBER CORRADINI: Plus a cool down?

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

5 MEMBER CORRADINI: Okay, thank you.

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

11 (Off microphone comment)

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

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

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

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

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

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

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

18 And the QHOS are the standard
19 complementary cumulative distribution function over
20 the average, from the EAB, one mile out for the
21 prompt, from the EAB, ten miles out for the latent.
22 Next page.

23 MEMBER ARMIJO: Could you just go back to
24 that?

25 MR. SILADY: Sure.

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

3 MR. SILADY: Yes?

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

7 MR. SILADY: Well, we hope --

8 MEMBER ARMIJO: From a plant standpoint?

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

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

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

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

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

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

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

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

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1 MEMBER ARMIJO: Yes.

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

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

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

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

23 We're going to look at internal and
24 external events, all operating modes, full scope PRA
25 basically. And it's going to, as we've already

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1 mentioned multiple times, have multiple reactors.
2 And it's going to look at nearby hazards as well.

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

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

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

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

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

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

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

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

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

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

25 In this particular event, there were

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

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

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

12 But when they fail they end up in the
13 design basis event region. So this is the path that
14 leads to an event below 10 of the minus 2, and above
15 10 of the minus 4.

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

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

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

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

25 MEMBER ARMIJO: Right, okay. Got it.

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

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

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

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

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

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

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1 that's active, I'll put two of those in.

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

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

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

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

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

23 And we said, well, we're going to put
24 into Chapter 3 and Chapter 15 the one, the RCCS that
25 we've designed with such reliability and capability,

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1 rather than rely on the one that might do it if this
2 fails. So this is the one we selected as safety
3 related. And that's our process.

4 MEMBER CORRADINI: So can I say it
5 differently?

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

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

15 MR. SILADY: That's correct.

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

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

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

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

25 MEMBER CORRADINI: I understand. But you

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1 do automatically.

2 MR. SILADY: This technology does.

3 MEMBER CORRADINI: Okay, fine.

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

6 MR. SILADY: It's technology-dependent.

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

9 MR. SILADY: Yes.

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

12 MR. SILADY: Yes.

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

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

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

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

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1 the consequences.

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

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

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

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

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

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

21 MR. SILADY: Yes.

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

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

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

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

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

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

17 MEMBER ARMIJO: Okay.

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

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

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1 you can stop me.

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

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

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

12 MEMBER SHACK: It's just a proposal.

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

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

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

25 DR. KRESS: When I think LWRs, they

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

5 MR. SILADY: That's correct. That's
6 intermediate metrics so they don't have to do a Level
7 3 all the way out.

8 DR. KRESS: Yes.

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

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

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

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

25 But the licensing basis events are the

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

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

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

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

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

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

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

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

17 We submitted, back in July of 2010, next
18 slide, we submitted a White Paper on the subject of
19 mechanistic source terms. The ADAMS number is there.
20 If we go to the next slide, that paper contained
21 information on radionuclide transport and retention
22 in modular HTGR's, a description of the functional
23 containment, a discussion of behavioral radionuclides
24 in the plant, mechanistic source term models and
25 modeling assumptions that have been used over the

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1 years in the modular HTGR business. Sources of data
2 on radionuclide behavior that are used in that model,
3 those models and the experimental methods for data
4 collection.

5 For the next slide our July 6th letter
6 requested NRC staff provide us some positions on
7 functional containment and on mechanistic source
8 terms. With regard to functional containment, we
9 asked the staff to establish some options regarding
10 functional containment performance standards for
11 modular HTGR's. And this is analogous to what was
12 requested by the Commission in the SRM, the SECY-03-
13 0047 and is further discussed in SECY-05-0006.

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

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

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

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

25 MR. ALBERSTEIN: In the --

1 CHAIR BLEY: Should I ask staff about
2 this?

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

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

15 CHAIR BLEY: Thank you.

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

23 There's a long history going back to the
24 late 80's and early 90's of regulatory staff and the
25 industry taking a look at these kind of issues. It's

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1 addressed in the Advanced Reactor Policy Statement.
2 It's addressed in a number of SECYs including some
3 recent ones for advanced LWRs that talk about new,
4 revised or physically based source terms.

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

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

23 And we're in a similar position today.
24 And that's pretty consistent, I think, with what I
25 just described in the February 15th working group

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1 assessment reports. In 1995, the new reg 1338 was
2 updated. And it was noted then again that a
3 mechanistic source specific to the design is an
4 acceptable approach.

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

13 DR. KRESS: Question.

14 MR. ALBERSTEIN: Yes.

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

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

25 DR. KRESS: I was wondering if there was

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

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

8 DR. KRESS: That would limit your power?

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

12 MEMBER CORRADINI: Power level or power
13 density?

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

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

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

20 DR. KRESS: Yes.

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

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

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1 problem with those.

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

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

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

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

21 Then and in the beyond design basis event
22 space, we looked at a more severe more, 0.7 g. And
23 it gets very difficult to put a frequency number on
24 it, site dependent and so on. But it was roughly in
25 the 10^{-6} range. And we assessed it, best estimate on

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1 the capability of the plant to respond having been
2 designed for 0.3 g and found that point on the chart
3 and treated it accordingly in terms of top level
4 regulatory criteria and so on.

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

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

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

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

23 MR. SILADY: Well, I think the Committee
24 understands that both on the frequency and consequent
25 space the per plant year is key. And we try to pick

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1 up events that effect all the modules. But then they
2 may each respond differently. Shut down cooling
3 system on one may survive the earthquake and on
4 others it may not.

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

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

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

24 DR. KRESS: Let me ask you a strange
25 question. If you have multiple modules do you just

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1 have one control room and one set of controllers?

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

4 MEMBER CORRADINI: Say that again.

5 MR. SILADY: Yes. One control.

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

8 MEMBER CORRADINI: No.

9 MEMBER SHACK: For the suite.

10 DR. KRESS: For the whole suite.

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

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

22 MR. SILADY: We have four on the MHTGR.

23 MR. ALBERSTEIN: The specifics there
24 would be subject to whatever the designer of the next
25 plant comes up with. Okay. Moving on with regard to

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1 containment alternatives in the staff's evaluation of
2 the modular HTGR in the 1995 update of the safety
3 evaluation, it was noted that Commission had already
4 noted in other proceedings that conventional LWR type
5 containments are not an absolute requirement for
6 advanced reactor designs.

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

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

23 Next slide. We refer in our White Papers
24 and in our presentations to something we call the
25 functional containment. What is this thing? What it

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1 is, it's a collection of design selections that taken
2 together ensure that the radionuclides are retained
3 at their source in the fuel for our safety design
4 strategy and that the regulatory requirements like 10
5 CFR 5034, 5279 for offsite dose and our plant design
6 goals of beating the PAG's are met at the Exclusion
7 Area Boundary.

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

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

24 Then the other barrier you have is the
25 vessel and the steam generator tubes and no

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1 containment. And the reactor building is not a
2 containment. And that's been an age-old issue
3 whether you really need a containment if you truly
4 have fuel with the characteristics of the TRISO fuel
5 and its operated at a low enough temperature.

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

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

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

20 MR. ALBERSTEIN: We'll get there. So
21 let's move on. Our definition, I already had it in
22 an earlier viewgraph, we define the source term as
23 the releases from the reactor building, the stuff
24 that doses the public taking into account the timing
25 which can for the most severe events consist of an

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1 initial release followed by a delayed release,
2 physical and chemical forms and the thermal energy.

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

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

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

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1 more of a manageable problem.

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

9 DR. KRESS: Ruthenium not in there?

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

12 DR. KRESS: You treat iodine as molecular
13 iodine?

14 MR. ALBERSTEIN: Yes.

15 DR. KRESS: I guess it has to be.

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

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

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

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

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

24 MR. ALBERSTEIN: Chromium, cesium,
25 silver, they're all --

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

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

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

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

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

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

22 But what this viewgraph is, is an attempt
23 to capture in one picture all of the phenomena that
24 take place with regard to fission product generation
25 and transport within the plant systems and that need

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1 to be taken into account when doing modeling of
2 source terms. Some of these phenomena occur only
3 during accidents. Not all of them occur during all
4 accidents.

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

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

20 Plate out the condensation of
21 radionuclides on those HPB surfaces is represented
22 down in the lower right. You see the effects of the
23 helium purification system which is used primarily to
24 ensure that residual oxidant levels in the helium are
25 held to acceptably low levels. But that also has the

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1 advantage of removing some circulating radionuclides.

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

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

25 MEMBER REMPE: How much uncertainty

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

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

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

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

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

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

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

1 MR. ALBERSTEIN: Has not been identified
2 where.

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

6 MR. ALBERSTEIN: Primarily.

7 MALE PARTICIPANT: This covers both
8 technologies, so --

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

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

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

16 DR. KRESS: From the pebbles.

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

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

25 MR. ALBERSTEIN: Well certainly the

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

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

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

11 MR. PETTI: Equilibrium core, right?

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

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

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

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

23 MR. ALBERSTEIN: Yes.

24 MR. PETTI: It's incredibly tedious.

25 MR. ALBERSTEIN: Let's move on to the

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

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

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

22 So we have a design target for
23 incremental release for an incremental fuel failure
24 during licensing basis events, about another 10^{-4} .
25 What we found is, in our analyses, is that

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1 radionuclide release during LBEs tends to be
2 dominated by the exposed heavy metal. The
3 contamination and the kernels that are exposed either
4 during the initial fabrication process or the
5 incremental exposure of kernels during normal
6 operation or the transience.

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

9 MR. ALBERSTEIN: Yes.

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

13 MR. PETTI: The specification is actually
14 double that $2 \cdot 10^{-5}$.

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

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

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

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

4 MR. ALBERSTEIN: You're in the weeds.

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

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

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

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

23 MEMBER ARMIJO: You do that.

24 MR. PETTI: There's all sorts of things
25 that are done, that's been learned over the years

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1 when they used to make 10^{-3} fuel. How do you get down
2 to 10^{-5} ?

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

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

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

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

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

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

25 MEMBER CORRADINI: Would you be able to

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1 a like a light water reactor determine where?

2 MR. PETTI: No.

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

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

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

12 MR. PETTI: That's an owner concern.

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

15 MR. ALBERSTEIN: Somewhat, yes.

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

20 MEMBER ARMIJO: I would guess.

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

24 MR. ALBERSTEIN: But in answer to your
25 question, at least for a prismatic core, pinpointing

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1 an exact fuel element that has fuel that's giving you
2 a problem is a problem. For pebbles, you pick it up
3 in the refueling machine. You can pick it up in the
4 process of circulating the pebbles. But for
5 prismatics that's an issue.

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

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

25 And I believe the next slide, yes, okay.

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

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

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

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1 to release of radionuclides from heavy metal
2 contamination.

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

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

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

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

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

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

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

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

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

7 MEMBER ARMIJO: Don't worry about that.

8 CHAIR BLEY: Just come up and join us.

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

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

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

21 MEMBER ARMIJO: So it was less than one
22 percent and your design goal was five --

23 MR. HANSON: Was five percent.

24 MEMBER ARMIJO: Okay. And then you're
25 dropping that down a couple orders of magnitude for

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1 the MHTGR, is that?

2 MR. HANSON: Yes.

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

10 The PBMR folks have other codes that do
11 analyses of these types of radionuclides. And again
12 you can see measured was less than calculated, a good
13 thing. In the case of condensable radionuclides the
14 metallic rate of nuclides typically at GA the
15 objective was to get them right within a factor of
16 ten. At PBMR it was a factor of five for some
17 isotopes, a factor of ten for others and a factor of
18 20 for others. GA tends to shoot, tended to shoot
19 just for a factor of ten and you can see the results
20 are within that range.

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

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

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

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

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

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

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

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

21 DR. KRESS: I understand.

22 MR. ALBERSTEIN: Okay. Let's move on.
23 Fred mentioned and I think I mentioned earlier, that
24 for off-normal events one can have release of
25 radionuclides in two phases. An early release and

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1 then a delayed release. Not all licensing basis
2 events entail a delayed release, but for those that
3 do we'll get into a little bit of the mechanisms.

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

13 For large breaks you get large shear
14 forces within the helium pressure boundary as the
15 helium is depressurized out through the breach. And
16 in those situations where the shear force on a given
17 surface, on a given location within the helium
18 pressure boundary becomes higher than the shear force
19 during normal operation, some of the condensed
20 radionuclides can be re-entrained and subsequently
21 released from the helium pressure boundary. Again,
22 the amount depends on the size of the break and
23 therefore on the size of the shear forces within the
24 helium pressure boundary and on the location.

25 For certain accident scenarios like a

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

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

18 This being an acceptable strategy hinges
19 upon being able to manufacture the fuel with low
20 levels of contamination, low levels of initially
21 defective fuel particles and operate the reactor with
22 low levels of incremental fuel failures such that
23 when one releases that vented release to the
24 environment, the objective of meeting the PAGs,
25 design objective of meeting the PAGs at the EAB and

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1 the regulatory requirement to meet 5034 of the EAB is
2 still met.

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

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

9 MR. ALBERSTEIN: The geometrical
10 configuration of?

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

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

16 MEMBER CORRADINI: Good.

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

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

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1 MR. ALBERSTEIN: Yes.

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

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

9 MEMBER CORRADINI: But I, okay, well.

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

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

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

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

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1 release.

2 MR. ALBERSTEIN: That's part of it.

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

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

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

12 MR. ALBERSTEIN: The reactor building
13 does provide some attenuation for radionuclide
14 release. We're going to jump ahead to the punch line
15 in one of my later slides. Analyses that have been
16 done so far indicate that relative to meeting the
17 regulatory requirements at the exclusionary boundary,
18 5034, 5279 --

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

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

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

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

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

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

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

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

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

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

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

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

15 MEMBER CORRADINI: Yes.

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

22 The no part is traditionally it's been
23 safety related for light water reactors to retain
24 radionuclides. We don't need it to do that. We do
25 need it to meet the PAGs. But we don't need it to

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1 meet 10 CFR 5034. So we want to put the focus on the
2 reactor building on what we, is necessary and
3 sufficient.

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

6 MR. ALBERSTEIN: That's correct. Yes.

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

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

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

18 Is it required for the NRC safety goals,
19 no. Is it required for the design basis events for
20 10CFR 5034, no. Is it required, you know, so we
21 classified it safety related. But we really want to
22 keep the focus and the effort on protecting of
23 external events and maintaining the core geometry.

24 MS. BANERJEE: Are we taking these as an
25 action item for the April 9th presentation then to go

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1 more over the defense in depth?

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

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

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

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

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

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1 understand what we're talking about here.

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

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

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

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

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

25 Historically we've found, not yet, we've

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1 found that for those accident scenarios which have a
2 delayed release and they tend to be the dose-dominant
3 scenarios, the delayed release is typically larger
4 then what you get from release of circulating
5 activity and any amount of liftoff or washoff. It
6 depends on how much time the fuel spends at
7 temperature.

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

16 And again, the delayed release is a
17 function of location and size of breach of the
18 primary system. And then the timing relative to the
19 heat up and cool down of the core. And as Fred
20 mentioned, a small leak actually has a greater
21 release, can have a greater release than a larger
22 release, then a larger breach from the helium
23 pressure boundary.

24 And I think we already touched on the
25 rest of these sub ticks, except the last one that

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1 once the temperatures within the helium pressure
2 boundary decrease as the core cools down, then the
3 releases will cease eventually as the core gets to
4 lower temperatures and there's no further driving
5 force to support the release.

6 The next slide is a representative
7 presentation of functional containment performance
8 during a depressurized loss of forced cooling event.
9 And what we're trying to show you here is the
10 relative effectiveness of each of the barriers of the
11 functional containment in the retention or
12 attenuation of fission products throughout the
13 functional containment.

14 So taking as an example the green bars
15 which show iodine production and release, the
16 particular analysis here was for the modular HTGR in
17 1989. You got 10 million curies roughly of iodine-
18 131 in the core to begin with. That which gets out
19 of the fuel is attenuated by about four quarters of
20 magnitude.

21 In the models it's assumed that whatever
22 gets out of the particles also gets out of the
23 graphite and into the circulating activity. So you
24 see no attenuation in the next step. And then that
25 which can get out of the, out of the graphite into

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1 the primary boundary under this particular accident
2 scenario, it's attenuated by another factor of about
3 20, by the helium pressure boundary itself.

4 And then you can see in the last step
5 there's another attenuation factor of about a factor
6 of 10, which is what's provided by the reactor
7 building.

8 MEMBER ARMIJO: And that's for a reactor
9 building that can exchange all the air in --

10 MR. ALBERSTEIN: That's for the reference
11 design.

12 MEMBER ARMIJO: The reference design. So
13 it's a leaky building.

14 MR. ALBERSTEIN: Compared to an LWR, yes.

15 MEMBER ARMIJO: Okay.

16 MR. ALBERSTEIN: On the other hand, for
17 cesium and strontium you can see again the retention
18 by the fuel particles is about the same. But the
19 amount that's attenuated by the matrix and the
20 graphite material differs, the retention factors
21 differ when you compare cesium to strontium.

22 But again overall we're talking about
23 retention of radionuclides by six to eight orders of
24 magnitude which is consistent with the safety design
25 approach of retaining radionuclides at their source.

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1 So a lot to chew on here. But that's a brief --

2 CHAIR BLEY: But just a couple
3 simpleminded questions from me if you would. The
4 charts, is this the result you're showing here
5 consistent with the experiments you've run in the
6 first two boxes I guess, the first two columns that
7 Dave will be talking about later?

8 MR. PETTI: No, these are higher.

9 CHAIR BLEY: These are higher. The basis
10 for these were?

11 MR. PETTI: Back in 1989, what they
12 thought --

13 CHAIR BLEY: Okay that's what this is
14 that we're looking at.

15 MR. PETTI: So we would show you that the
16 release from fuel is even better.

17 CHAIR BLEY: And would there be any
18 difference if for the next step if you had pebble bed
19 or if you had prismatic? Do we know?

20 MR. ALBERSTEIN: I don't think I could
21 speak to that.

22 MR. PETTI: Which step?

23 CHAIR BLEY: Released from the graphite.

24 MR. PETTI: Yes. That's complicated.

25 There's a little bit and there's some canceling

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1 factors. I think it's in the same order of
2 magnitude.

3 MEMBER ARMIJO: Will you be showing us a
4 chart, the same kind of chart with modern fuel, fuel
5 you've been testing?

6 MR. PETTI: We should have been. No,
7 what I will show you though is relative to the
8 requirements. So you saw some discussion on
9 incremental failure. I'll show you what that means.
10 So it won't look at the curies. But there will be
11 some relative ratios that will translate directly.

12 CHAIR BLEY: And this is a depressurized
13 loss so when you get to the last two boxes getting
14 out of the reactor building there's no driving force
15 it's just air circulation?

16 MR. PETTI: Well there's the initial
17 release that's in --

18 MR. SILADY: But that's not dominant
19 here. And you're right there's, in fact it depends
20 on the timing. It may be sucking nuclides back in to
21 the helium pressure boundary.

22 CHAIR BLEY: Go ahead. I was just trying
23 to figure out what we were really looking at here.

24 MR. ALBERSTEIN: There's a lot of data
25 here. And we were trying to find a relatively

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1 succinct way to present it. Move on to the next
2 slide. In summary on functional containment our
3 emphasis again is on radionuclide retention within
4 the fuel during normal operation with the release of
5 a relatively low inventory of radionuclides to the
6 helium pressure boundary.

7 The limiting off-normal events tend to be
8 characterized by an initial release from the helium
9 pressure boundary that's a function of leak size,
10 break size, pressure relief performance and so on and
11 so forth. And then a larger delayed release from the
12 fuel.

13 Our analyses thus far have indicated that
14 this functional containment, this overall system of
15 barriers, will meet the regulatory requirements for
16 offsite dose at the EAB with margin for a wide
17 spectrum of off-normal events without even taking
18 into account the retention factors of the reactor
19 building. But to meet the EPA PAGs at the EAB with
20 margin, we do need to take into account the retention
21 of radionuclides by the reactor building. So moving
22 on.

23 MEMBER CORRADINI: So, I'm sorry that I
24 have to get pulled out. But just let me repeat what
25 you said before and make sure I didn't mishear. So

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1 with your last, your fourth bullet the reactor
2 building is filtered or is not filtered in your
3 current design concept?

4 MR. ALBERSTEIN: It is not.

5 MEMBER CORRADINI: And Fort St. Vrain was
6 not also?

7 MR. ALBERSTEIN: It was not.

8 MEMBER CORRADINI: Okay. I guess my
9 memory banks say it was but, okay.

10 (Crosstalk)

11 MEMBER REMPE: It had a what?

12 MR. PETTI: An HVAC, heating, ventilating
13 and air conditioning but it wasn't available in DVA
14 number one.

15 MEMBER CORRADINI: Okay. So it was there
16 but it wasn't called upon in the analysis?

17 MR. PETTI: We will have heating,
18 ventilating and air conditioning in the reactor
19 building too.

20 MR. ALBERSTEIN: But it's not --

21 MR. PETTI: But if you go to the
22 frequencies of these things, it often times isn't
23 there.

24 MR. ALBERSTEIN: And I guess on this
25 point of whether the reactor building is filtered or

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1 not I think what we're trying to present here is the
2 process and the definition for a mechanistic source
3 term.

4 MEMBER CORRADINI: I'm with you.

5 MR. ALBERSTEIN: I think we recognize
6 that the reactor building is one of the attenuators
7 of a release. And it will be in the details of the
8 final design as to whether the --

9 MEMBER CORRADINI: I understand but just
10 so I remember it though, the attenuation is occurring
11 by physical processes without a filter?

12 MR. ALBERSTEIN: Yes. There have been
13 various different designs over the decades, some have
14 had filters.

15 MEMBER CORRADINI: I just wanted to
16 verify that.

17 MR. PETTI: That's largely the longer the
18 delayed release is what's getting attenuated. That
19 initial venting, there's nothing that, you know, it's
20 not in there long enough, so.

21 CHAIR BLEY: But you haven't been heated
22 up.

23 MR. PETTI: Right.

24 MEMBER ARMIJO: Just before you leave
25 this, when you say at the last bullet you can meet

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1 these requirements for a wide spectrum of off-normal
2 events does that mean all the identified off-normal
3 events that you're designing for? What is the one
4 that's not, you're not capable of providing this
5 meeting of PAGs?

6 MR. ALBERSTEIN: We've not yet found it.
7 But we will in the next presentation, talk about
8 bounding event sequences which go beyond design
9 basis.

10 MR. SILADY: To clarify though it's the
11 design basis events and the beyond design basis
12 events meeting the PAGs.

13 MEMBER SHACK: But it's with your current
14 definitions of what a beyond design basis event is?

15 MR. SILADY: Correct.

16 MEMBER ARMIJO: Got it.

17 MR. ALBERSTEIN: Okay. I think I have
18 only one more slide, well don't count that one. I
19 have one more slide. And this is sort of our overall
20 conclusions. Number one, we believe that the
21 approach to functional containment and mechanistic
22 source term being taken for modular HTGRs is
23 consistent with the Advanced Reactor Policy
24 Statement.

25 It's consistent with discussions of

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1 containment function and mechanistic source terms in
2 a wide variety of SECY documents that have been
3 issued over the last 20 years. And it's consistent
4 with approaches that have been previously reviewed by
5 the NRC staff for modular HTGRs, particularly for the
6 MHTGR reviews in the 80's and 90's. It's also
7 consistent with the approaches in the pebble bed
8 reviews that were done roughly ten years ago.

9 We take an event specific approach that
10 can be applied to the full range of licensing basis
11 events using mechanistic models for fission product
12 generation and transport accounting for the inherent
13 behavior of HTGRs, their passive design features and
14 the mechanical performance of the fission product
15 release barriers that comprise the functional
16 containment. And that's all I have on that topic.

17 MEMBER RAY: You got one more slide
18 according to this.

19 MR. ALBERSTEIN: Do I?

20 MR. PETTI: I did too and I missed it.
21 I set the pattern here.

22 MR. ALBERSTEIN: Okay. We'll get this
23 right the next presentation. This is just a summary
24 recap of the things we've requested the NRC staff to
25 give us positions on. Shorthand summary of the ones

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1 that were in the earlier viewgraph. So now I'm done.

2 CHAIR BLEY: And now you have another.

3 MR. ALBERSTEIN: Yes, the fun doesn't
4 stop. As I said at the beginning of this
5 presentation, the next presentation, this
6 presentation on siting source terms in somewhat a
7 subset or a specialized aspect of the overall topic
8 of miscellaneous source terms.

9 So if we can move to the next slide.
10 That's where we are in the agenda. And next slide.
11 I'm going to talk a little bit about the staff
12 position regarding site and source terms that we've
13 requested. We're going to talk about the approach to
14 be taken to siting source terms.

15 Then we're going to talk about a further
16 specialized aspect of this which is event sequences
17 involving graphite oxidation. We're going to get
18 into that because it's been the subject of a lot of
19 discussion between the project and the staff over the
20 last several months. So we felt we should address it
21 here. And then we'll give you some conclusions
22 overall on siting source terms.

23 CHAIR BLEY: For the issues you're going
24 to talk about here, they're not covered in your
25 mechanistic source term White Paper, are they or are

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1 they embedded in there?

2 MR. ALBERSTEIN: They're really somewhat
3 separate.

4 CHAIR BLEY: You don't have a White Paper
5 on this one?

6 MR. ALBERSTEIN: No, we don't. These are
7 issues that have come up since the staff issued its
8 working group assessment report on February 15th.
9 And as I've said, we've spent a lot of time on this
10 with the staff. And it's garnered quite a bit of
11 attention and that's why we wanted to give you a
12 presentation on it here today.

13 So requested position, next viewgraph.
14 A lot of words here. But siting source terms, in the
15 light water reactor community, are developed based on
16 an assumption that one looks at an accident sequence
17 that entails a substantial meltdown of the core with
18 subsequent release of pretty large quantities of
19 fission products. That comes up in the footnotes to
20 10 CFR 5034, 5279 and one of the subsections of 10
21 CFR 100 in the earlier days of the regulations.

22 And taking that language that talks about
23 melting of the core and applying it to a reactor that
24 number one has no metal in the core, number two has
25 taken a safety design approach to ensure that relying

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1 only on passive behavior of the plant, the fuel isn't
2 going to reach temperatures at which significant fuel
3 particle coating failure can occur. That creates for
4 us a bit of a dilemma.

5 How does one address this regulatory
6 requirement in the context of an HTGR? So what we
7 requested is that the staff develop a position to
8 give us some final determination of regarding how
9 licensing basis events would be considered for the
10 purpose of plant siting and functional containment
11 design decisions. Taking into account that the staff
12 has previously found that improved fuel performance
13 is a justification to revise siting source terms and
14 containment design requirements.

15 Next slide. So what approach are we
16 going to take here? Well the approach that we plan
17 to take is patterned after that which was developed
18 back in the late 80's and the early 90's in the
19 modular HTGR review. This approach was documented
20 both in the PSID and the PRA for that reactor. And
21 the findings regarding their approach were discussed
22 in the staff safety evaluation NUREG-1338.

23 So the first step is to develop the
24 design consistent with the safety design approach
25 that Fred has already described and to utilize risk

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1 insights as input to design, to the design for the
2 range of requirements, both user requirements and
3 regulatory. And then select and mechanistically
4 evaluated LBEs including the DBEs, DBAs and BDBEs
5 against the top level regulatory requirement and
6 against our design goal of beating the PAGs at the
7 EAB.

8 And I'm consistent with the mechanistic
9 source term approach do a mechanistic evaluation of
10 these events that have limiting dose consequences,
11 the highest dose consequences offsite and use those
12 source terms as the siting source terms. Go to the
13 next slide, give you a little more information.

14 Fred already showed you that for the
15 MHTGR they identified three design basis accidents
16 that were the highest offsite dose consequence with
17 the limiting DBAs. And Fred's already gone through
18 the brief description of each of these. It's
19 interesting to note that each of these entails
20 ingress of either moisture or air into the reactor.
21 So if one were doing this for that old design, these
22 would be the limiting design basis accidents that
23 would be used to generate siting source terms.

24 MEMBER RAY: This is a single tube
25 rupture in six, is it?

1 MR. ALBERSTEIN: Correct. That was a
2 single tube rupture. And we're going to talk about
3 a scenario with more tubes in just a second. We can
4 go on to the next slide.

5 During the review of the modular HTGR by
6 the staff back in the 80's and 90's, they started
7 asking a series of questions about, I think the
8 attempt here was to try to determine what kind of
9 margin one has in this plant given that approach to
10 selecting limiting design basis accidents. And the
11 staff postulated a number of bounding event
12 sequences, and we'll show them to you in just a
13 second here, to try to test just how far the plant
14 could be pushed while still meeting the regulatory
15 requirements and the design goals in the way that GA
16 and DOE were attempting to do at that time.

17 So we would take elements of that from
18 that review and use them in siting source term
19 determination today. Specifically what we would do
20 is that to ensure that there aren't any cliff edge
21 effects out there where things could go bad
22 unexpectedly and to understand just how much margin
23 we have, we would supplement the LBE-derived siting
24 source terms with insights from a best estimate
25 mechanistic evaluation of some bounding event

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1 sequences.

2 Don Carlson has referred to this as a
3 survey of the safety terrain, just to see what's out
4 there. And I think that's a good way to describe it.
5 But this isn't a free for all. This isn't an
6 opportunity for people to exercise their imagination
7 and come up with exotic scenarios that don't make any
8 sense.

9 Number one, they need to be physically
10 plausible rather than just non-physical, arbitrary
11 combinations of event parameters. Now physically
12 plausible is a subjective term and it's intentionally
13 subjective here. It means you don't pretend that the
14 laws of physics have suddenly been suspended in order
15 to come up with some exotic accident scenario.

16 It means that you don't suddenly assume
17 that the physical properties of the materials in the
18 core are radically different from what they're known
19 to be just to try to create some kind of large
20 release. They have to be physically plausible. They
21 have to be sensible. And we'll give you some
22 examples in a minute here.

23 These are event sequences that are in 10
24 to the minus double digit frequency range. And it's
25 pretty hard to rigorously quantify frequencies when

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1 you get down to those kinds of numbers. But
2 nonetheless we do expect that generally these kinds
3 of sequences would have frequencies lower than the
4 BDBE region which has it as it's minimum cut off 5
5 times 10^{-7} for plant year and frequency.

6 As we evaluate the events we're going to
7 consider again the intrinsic and passive behaviors of
8 the HTGR. Next slide. So how do you determine what
9 the bounding event sequences would be? You do a
10 deterministic and this is a purely deterministic
11 process, by the way. You do a deterministic review
12 of plausible events that potentially impact the
13 safety functions of removing core heat, controlling
14 heat generation and controlling graphite, controlling
15 chemical attack, for example graphite oxidation.

16 In order to do an initial selection of
17 bounding event sequences you have to have your design
18 fairly well established. You have to be through
19 preliminary design. But we can say at this time that
20 the bounding event selection process we'd use as a
21 starting point. The six bounding event sequences
22 that were requested by the NRC staff back in its
23 review of the MHTGR.

24 And if we go to the next viewgraph you'll
25 see what those were. And I'm just, you can read

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1 these for yourself. I'm going to highlight a couple
2 of them okay.

3 MEMBER CORRADINI: These are the ones
4 that the staff back in '89 asked you guys to
5 consider?

6 MR. ALBERSTEIN: Yes. That's what they
7 are. One of them was in inadvertent withdrawal of
8 all control rods without scram for a 36 hour period.
9 And the 36 hours is the number that the staff came up
10 with at that time. I think it was a reflection of
11 their thinking at that time that within 36 hours some
12 kind of action would be taken to mitigate the
13 consequences of a sequence like this.

14 I know that in today's world we're
15 talking about different lengths of time, longer
16 lengths of time. That's okay. But at the time this
17 was what they were working with. So inadvertent
18 withdrawal of all control rods is one example.

19 Number four, steam generator tube rupture
20 that takes out 25 percent of the tubes with failure
21 to isolate or dump. Number five, a rapid
22 depressurization of one module resulting from a
23 double-ended guillotine break of what they call the
24 crossduct, it's actually a Section 3 cross vessel
25 with a failure to scram and an assumption that the

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1 reactor cavity cooling system is failed or
2 unavailable for 36 hours. And thereafter only 25
3 percent of it is unblocked.

4 So these are pretty bounding, severe
5 types of events, okay. They all resulted in low
6 offsite doses. They all, when GA did its exit
7 accident analyses for these sequences they still
8 resulted in the ability to meet the PAGs at the
9 exclusionary boundary.

10 So this gives you some sense, that gives
11 you some sense of, that there's quite a bit of margin
12 here. In terms of frequency assessment, all most all
13 of these were in the ten to the minus double digits
14 regime, which one would expect also.

15 So we would use these as a starting
16 point. But whatever we eventually choose as the
17 bounding event sequences, we can go to the next
18 slide, but what would we do with the results of the
19 analyses? The applications would be number one,
20 they'd be used to identify and understand the
21 potential for cliff edge effects, for high
22 consequence events.

23 Theoretically, we may find something. We
24 may find something that has more risk than we thought
25 we did, that we had in this. There might be

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1 something out there. And so it's an opportunity to
2 determine potential risk significant plant or system
3 vulnerabilities. And if we do find a cliff out there
4 to identify risk mitigation strategies, most likely
5 design changes, that we would have to implement to
6 address that.

7 And the result of all of this is going to
8 be documented as part of the license application
9 process. It's not clear where in the structure it
10 would go, whether it would be topical reports,
11 Chapter 19, it's really not Chapter 15 material. But
12 it would get documented as part of the record.

13 There have been some previous staff
14 positions taken on bounding event sequences. In the
15 1989 version of the draft safety evaluation, the
16 staff indicated that it judged that these bounding
17 event sequences they had proposed, the results of
18 those analyses showed that the MHTGR had the
19 potential to cope with these rare, severe events
20 without the release of a significant amount of
21 fission products.

22 The ACRS also in the safety evaluation
23 back in those days, noted that neither the designers
24 nor the staff or the ACRS members themselves had been
25 able to postulate any accident scenarios of

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1 reasonable credibility for which additional physical
2 barriers to the release of fission products would be
3 required to provide adequate protection of the
4 public. The additional barriers might include a
5 filter on a vent. At least at that time, there was
6 no need identified.

7 Move on to the next slide. I'm going to
8 talk a little bit about graphite oxidation event
9 sequences. And the reason we want to talk about this
10 is that back in 1993 the Commission issued a staff
11 requirements memo stating that the Commission
12 believed that for the MHTGR the staff should be
13 addressing an event entailing the loss of primary
14 coolant pressure boundary integrity whereby ingress
15 could occur from the so-called chimney effect and
16 we'll talk about what that means, resulting in a
17 graphite fire and the subsequent loss of integrity of
18 fuel particle coatings.

19 You'd have to oxidize a lot of graphite
20 for that to happen. But the staff at that time
21 believed or somebody on the Commission at that time
22 believed that was a scenario that should be looked
23 at. And we've had quite a bit of discussion with the
24 staff over the last ten months, 11 months or so about
25 what should be done to address this old SRM.

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1 There are previous staff positions that
2 have been taken with regard to oxidation event
3 sequences in the modular HTGR. In the draft NUREG-
4 1338, it reflected the results of some independent
5 analyses that Brookhaven had done in support of the
6 safety evaluation. They noted that for graphite
7 oxidation to proceed to a point that structural
8 damage inside the core could be possible, you'd have
9 to have an unlimited supply of air for many days.

10 You'll recall from comments Fred made
11 earlier, for these depressurization events that do
12 result in some air ingress, it's not pure air that
13 gets in there. It's a mixture of air and helium and
14 it is in fact mostly helium. So Brookhaven concluded
15 that you'd have to have an unlimited air supply for
16 many, many days. And in the 1995 update to the
17 safety evaluation, the staff concluded that a
18 graphite fire in the MHTGR is a very low probability
19 event.

20 They also noted that as stated in another
21 NUREG done by one of their contractors, without two
22 breaches of the reactor vessel to create a chimney
23 effect, one up high and one down low, it's not likely
24 that significant amounts of air will enter the core
25 and therefore that graphite fires are not a

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1 licensability issue for the modular HTGR. To get a
2 chimney that's going to really move the air through
3 the core, you've got to have a breach of an ASME
4 Class 3 vessel at the top and you've got to have one
5 down low.

6 And I think all of you guys who are light
7 water reactor people will recognize that's a scenario
8 that goes far beyond the types of scenarios that have
9 typically been required in reactor safety regulation.
10 Onto the next slide.

11 MEMBER ARMIJO: You don't have any kind
12 of penetrations at the bottom of that vessel?

13 MR. ALBERSTEIN: At the bottom.

14 MEMBER CORRADINI: No, they have them in
15 the crossvessel.

16 MEMBER ARMIJO: Just the crossvessel. I
17 mean but really down at the bottom.

18 MR. ALBERSTEIN: At the bottom it's a
19 shut down cooling circulator down there.

20 MEMBER ARMIJO: Okay. That's the only --

21 MR. ALBERSTEIN: Yes. Go on to the next
22 slide. So what's the approach that we would take
23 today to event sequences involving graphite
24 oxidation? First of all consistent with the findings
25 of the staff in NUREG-1338 and with the findings of

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1 the ACRS back in the 90's, we really think that the
2 frequency of the kind of event described in the SRM
3 is going to fall so far below the LBE spectrum that
4 the event would be considered incredible. We think
5 this is a ten to the minus double digit type of
6 event.

7 Those expectations have to be confirmed
8 for the specific design of the next modular HTGR.
9 And that will be done. However, we recognize that
10 bounding event sequences that maximize the potential
11 for graphite oxidation do need to be considered, even
12 if it's not that particular scenario from the SRM.
13 And it's the intention to consider those in the
14 bounding event sequence process as part of the NGNP
15 design and licensing effort.

16 There are data needs in the area of both
17 air and moisture effects on core materials. When
18 Dave gives his presentation here he'll talk a little
19 bit about our plans in the AGR fuel development and
20 qualification program to address the effects and
21 obtain additional data on the effects of air and
22 moisture ingress.

23 Next slide. Conclusions, next slide.
24 Number one, the approach we're going to take is
25 essentially the same for siting source terms,

1 essentially the same as that proposed in the days of
2 the modular HTGR review in the late 80's and early
3 90's. I believe this approach is consistent with
4 discussions of containment function and mechanistic
5 source terms and more recent SECY documents and what
6 approach is previously reviewed by the staff.

7 Limiting LBEs will be evaluated to determine SSTs and
8 physically plausible bounding event sequences,
9 including some involving graphite oxidation will be
10 considered to make sure there are no cliff edges.

11 DR. KRESS: What exactly does that mean?

12 MR. ALBERSTEIN: Pardon.

13 DR. KRESS: I'm not sure exactly what
14 that means because if you're going to include a
15 graphite fire.

16 MR. ALBERSTEIN: We will look at
17 sequences that entail graphite oxidation. You see
18 I'm judiciously avoiding the use of the f-word.

19 DR. KRESS: I don't mind. But you're
20 going to look at it and, you know, depending on how
21 much air you get in there it could have devastating
22 effects. So are you going to look at it, from what
23 standpoint? Limiting the amount of air or?

24 MR. ALBERSTEIN: We'll look at what we
25 believe are bounding ultralow frequency events that

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1 entail air ingress to assess their effects on the
2 performance of the system under such accident
3 conditions.

4 DR. KRESS: I'm still not sure what
5 you're going to do.

6 MR. SILADY: We'll probably do what we
7 did with the MHTGR.

8 DR. KRESS: I assume it's such a low
9 frequency.

10 MR. SILADY: No, we looked at, maybe we
11 can go to the backup slide, Mark, on graphite
12 oxidation if it's easy. Otherwise I'll just do it
13 verbally. He's going to pull it up. And I think if
14 you see it as well as I say it, there's a better
15 chance of communication.

16 It's number twelve. That's it. So some
17 of these things we've already talked about. Graphite
18 will oxidize with the oxygen in the air or in a
19 helium/air gas mixture. The nuclear grade graphite
20 is much less reactive than other types of graphite
21 due to its graphitized structure and high purity.
22 The oxidation of the graphite is limited by the
23 amount of air in the helium gas mixture from the
24 reactor building.

25 And then once that mixture comes in the

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1 high flow resistor in the coolant channels to the
2 core height, talking prismatic here, has an L/D
3 greater than 700. So it's hard to get air to go up.
4 It's going to react with these core support posts if
5 it's at the bottom or wherever it comes in. Fuel
6 particles are embedded in the graphite matrix. We've
7 talked about that.

8 And those little compacts are within the
9 fuel element. See you have to oxidize away a lot of
10 graphite to even get to the fuel particles which of
11 course have a silicon carbide layer on it. Loss of
12 all forced cooling and depressurization of the helium
13 pressure boundary are required for air to get, to
14 ingress to begin with in the mixture.

15 Sometimes we forget that. You've got all
16 these days and you have to not turn on any forced
17 cooling and cool the core down. And you have to have
18 a leak or a break of some size in the helium pressure
19 boundary. The chimney effect was mentioned in the
20 SRM. I suspect that if you have a really large
21 opening you get stratified flow as well.

22 But the point is, it is a very large
23 opening and you've lost forced cooling. Maybe those
24 are synonymous if it's that large. We did some
25 analyses. This is what we would probably do. This

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1 answers your question, what are you going to do?
2 Well when we looked at a 22 square foot opening at
3 the bottom of the vessel system, we didn't do a
4 double- ended guillotine break. But we did the size
5 of a vessel at the bottom of the vessel system. And
6 we only got one percent of the core graphite oxidized
7 after 30 days.

8 And that amount of air is equivalent to
9 eight reactor building volumes of a hundred percent
10 air. That's what we did in the 80's. And the reason
11 it was only one percent of the core graphite oxidized
12 is because we didn't get any additional decay heats
13 or heat generation from the exothermic reaction. But
14 the oxidation did not lead to a loss of core
15 geometry. And it was limited such that we didn't
16 incremental radionuclide release due to oxidation.

17 This by the way was two orders of
18 magnitude at one percent. We got 10^{-4} fraction from
19 design basis events. That was going factor of a
20 hundred greater. More recently NGNP analyses have
21 shown that a break in the helium pressure boundary
22 leads to a very small percentage of air in the gas
23 mixture after the helium blowdown.

24 So not only did it take 22 square feet,
25 not only would it take eight reactor building volume,

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1 so not only did we look at it for 30 days, but we
2 were assuming a hundred percent air.

3 Those analyses on a different reactor
4 building should have been more like two percent air
5 and 98 percent helium. We'll have to do this same
6 sort of thing, but we'll be fighting this for a long
7 time I'm sure. But we have to change the perceptions
8 of whoever's writing the SRMs.

9 MEMBER ARMIJO: Fred, where did the 22
10 square feet come from? What did you have to do to
11 create that kind of a --

12 MR. SILADY: We had to fail the vessel
13 either more plausibly it would probably be around the
14 weld of the cross vessel to the reactor vessel or the
15 cross vessel to the steam generator vessel. That was
16 it, that stratified flow.

17 MEMBER ARMIJO: Well did you ever fail
18 the penetration at the bottom of the post?

19 MR. SILADY: No, we didn't.

20 MEMBER ARMIJO: I think that would be
21 more likely than the cross vessels.

22 (Cross talk)

23 MEMBER RAY: Are the core support posts
24 too low in temperature to be concerned about?

25 MR. SILADY: No, they can oxidize.

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1 That's where most of the oxidation takes place. If
2 you get two moles for every mole. And so it sort of
3 chokes itself at some point trying to get up through
4 that cooling hose because the hot wants to rise.

5 MEMBER RAY: But it doesn't threaten the
6 function of --

7 MR. SILADY: We didn't find that we lost
8 any posts and the core is latticed such that one post
9 doesn't cause the core geometry anyway.

10 MEMBER RAY: I've got to go to another
11 meeting and so do you.

12 CHAIR BLEY: Lots of us have to go to
13 another meeting in just a couple of minutes.

14 MEMBER REMPE: Quick question. Earlier
15 you talked about the seismic analysis. And in many
16 cases you can rely on the various data. But are
17 there some specific components that are HTGR specific
18 and are there data for those HTGR specific
19 components?

20 MR. SILADY: With the response to a
21 seismic event?

22 MEMBER REMPE: Right. To give insights
23 on how you've quantified your seismic analysis.

24 MR. SILADY: I think a lot of the
25 structures are unique. I mean they don't have

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1 graphite core support posts.

2 MEMBER REMPE: And they never did any
3 sort of seismic, so is that another data need that's
4 been identified in your documentation?

5 MR. SILADY: Well this event, it's in the
6 0.3g sited source, safe shut down earthquake. It's
7 designed for that with margins. And then we looked
8 at it for a 0.7g which is more a 10^{-6} level. And
9 looked to make sure it had the capability with its
10 embedded below grade damped configuration.

11 MEMBER REMPE: What about a prismatic
12 fuel element? Anyone ever tested that to see what
13 happens with vibrations is kind of what I'm kind of
14 getting to? I mean are there a lot of things, are
15 there any data to help justify. You can design for
16 it, but --

17 MR. ALBERSTEIN: As I recall, it's a long
18 time ago, but as I recall there was a seismic
19 response testing program done for Fort St. Vrain
20 where they did some shaking of simulated smaller
21 versions of an HTGR prismatic core.

22 MR. SILADY: The Japanese did some of
23 this too.

24 MEMBER REMPE: That's what I was
25 wondering. I didn't know how much specific data

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1 there was.

2 MR. ALBERSTEIN: There's some information
3 out there.

4 MR. SILADY: I think it was in the PBMR
5 plans as well.

6 CHAIR BLEY: Okay. Do you have any more
7 you want to do or is that it?

8 MR. ALBERSTEIN: Was there one, the only
9 slide that was left was a recap.

10 CHAIR BLEY: Okay. I don't think we need
11 that. We're going to have to break for some of us to
12 go to another meeting. We'll all be back here at one
13 waiting to hear from Dave. That sounds pretty
14 interesting. So we're recessed until 1 o'clock.

15 (Whereupon, the foregoing matter went off
16 the record at 12:13 p.m. and went back on the record
17 at 1:01 p.m.)

18 CHAIR BLEY: The meeting is back in
19 session. And I'll turn it over to Dave Petti. Is
20 that right?

21 MR. PETTI: Yes. We'll talk about --

22 CHAIR BLEY: It's been a while since
23 you've actually been here, quite a while.

24 MR. PETTI: Yes, yes, yes. So --

25 CHAIR BLEY: Welcome back.

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1 MR. PETTI: Yes, we'll talk about where
2 we are on fuel qualification and fission product
3 behavior with a snapshot of the program. Next slide.
4 And yes, we are the last item on the agenda.

5 So I'll talk about what the White Paper,
6 the requested NRC staff positions are. We'll go to
7 background real quickly, talk about our approach to
8 qualification. And I pose a number of simple, key
9 questions that will help focus us as we go through
10 the presentation and then talk about the program plan
11 status and the key results, particularly as they
12 relate to licensing.

13 So this is not sort of my typical fuel
14 talk. This is sort of inside out instead of outside
15 in maybe. So it's to look at it from the licensing
16 sort of perspective. And I'll go through each of the
17 pieces of the program and talk about what it means
18 for licensing, what we've learned, where I think
19 we're going to end up because we're still in service
20 and then sort of a summary and what's it look like in
21 the future.

22 So the White Paper shown there is
23 submitted in July 2010. And the staff position is to
24 confirm that the plans being implemented by the
25 program are generally acceptable, provide reasonable

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1 assurance of the capability of the coated particle
2 fuel to retain fission products control in a
3 predictable manner. And particularly to identify any
4 additional information or testing needed to provide
5 adequate assurance of this capability.

6 So what we're really looking for is there
7 some big multi-million dollar, multi-year thing we're
8 missing? You know, little things you can incorporate
9 along the way. But if there's something big, we
10 really want to know now because this stuff, this work
11 you know, takes a long time and a lot of money.

12 MEMBER ARMIJO: Dave, before we go on I'm
13 going to have to leave early and there's one burning
14 question I want to leave with you and you can answer
15 whenever it's appropriate. I've gone through your
16 White Paper and a lot of the stuff you've done, fine
17 work that the laboratory has done. But fundamentally
18 you fabricate this fuel with batch processes, maybe
19 they're large batches I don't know.

20 But it's a batch process. And you're
21 talking about hundreds of thousands, maybe millions
22 of fuel particles to make up a core. And so it would
23 be multiple batches. And the question I want to ask
24 is
25 what is the NDT or inspection technique that assures

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1 you that each batch is at the same quality standards
2 when, you know, if a batch had 10 thousand failed
3 particles in it I would suspect that wouldn't meet
4 your criteria. But if the batch is 500 thousand
5 particles, how do you know? So somewhere along the
6 line just tell us how you ensure that you --

7 MR. PETTI: When are you leaving?

8 MEMBER ARMIJO: I have to leave at two.

9 MR. PETTI: I think we'll get there.

10 MEMBER CORRADINI: If we limit our
11 questions.

12 MEMBER ARMIJO: That's my one question.

13 MR. PETTI: Next slide. I think you all
14 know what the fuel looks like. Comes in, what I call
15 two flavors, either compacts or pebbles. We've been
16 focused heavily on compacts largely because when we
17 started this program over a decade ago there was very
18 healthy programs in pebbles internationally in China,
19 South Africa and Europe.

20 Also just a factor for consideration,
21 testing compacts is physically easier, they're
22 smaller items. Pebbles are big. It can be done but
23 it really limits what you can do. So next slide.

24 I think if you haven't go this message by
25 now from the previous presentations it's the fuel,

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1 it's the fuel, it's the fuel. We have to demonstrate
2 that we can retain the fission products and the key
3 principles are that we can make high quality low
4 defect fuel and characterize it in a repeatable,
5 consistent manner.

6 And so we'll talk about that. And that
7 the performance with very low in-service failure
8 rates is achievable within the envelope, the
9 operating envelope and the accident envelope and that
10 we can calculate that performance to the requisite
11 level of accuracy.

12 We are using a UCO which is a shorthand.
13 UCO, uranium oxycarbide is a mixture of uranium
14 dioxide and uranium carbide, both UC and UC₂ are
15 acceptable. This enables better performance at
16 higher burnup than UO₂. And particularly it
17 suppresses a failure mechanism in UO₂ known as kernel
18 migration where the kernel moves and can potentially
19 threaten the coatings. Because of the thermal
20 gradients this is more important in prismatics
21 because there are bigger gradients there.

22 You don't get any carbon monoxide
23 formation. Chemically you gather the free oxygen
24 produced from fission by the carbide phase so that
25 when you fission you free up that oxygen and it

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1 reacts with uranium carbide to form more UO₂ instead
2 of reacting with the carbon buffer to form carbon
3 monoxide which is what happens to UO₂.

4 And so the internal gas pressures are
5 reduced relative to the UO₂ TRISO that the Germans
6 and the rest of the world is looking at. And the
7 fission products are largely immobilized as oxides.
8 This is an engineered fuel form to tie up the fission
9 products largely as oxides. And you can get longer
10 more economical fuel cycles.

11 And just as a hint, I don't think I have
12 it is because of the work that we've done the rest of
13 the world is starting to look at UCO.

14 DR. KRESS: Quick question. UCO is a
15 mixture of these. What percentage of each in this
16 mixture?

17 MR. PETTI: Twenty-five percent carbide,
18 I believe, 75 percent oxide. But we have a pretty,
19 you don't, you have a good range. You don't have to
20 hit it on the dot. You just need --

21 DR. KRESS: It doesn't have to be that
22 precise.

23 MR. PETTI: It doesn't have to be that
24 precise, no. But so the exciting thing for us is
25 that the work that's being done here, people always

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1 thought UCO was very far off. And in fact it is now
2 being looked at by both the Koreans and the Chinese,
3 good performance because they see what it can do in
4 terms of the economics of the system.

5 DR. KRESS: Chinese have an operating
6 reactor?

7 MR. PETTI: They have a little ten
8 megawatt and they're building, they just put concrete
9 in their 250 megawatt pebble bed.

10 MEMBER CORRADINI: Where is that going to
11 be located? Is it at INET just north of Beijing?

12 MR. PETTI: No, it is a separate --

13 MEMBER CORRADINI: It's on the coast
14 somewhere?

15 MR. PETTI: It's on the coast somewhere.

16 MR. ALBERSTEIN: Twin unit.

17 MR. PETTI: Yes, it's a twin unit. So
18 our approach to qualification establish a spec. We
19 have specifications on the kernels, on the coatings
20 and on the compacts. We implement a process capable
21 of meeting that spec and implement statistical
22 quality control procedures to demonstrate the spec is
23 met. Unlike our LWR fuel we are not measuring on
24 every particle, obviously.

25 DR. KRESS: What are your specs on the

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1 coatings?

2 MR. PETTI: I'll, we'll talk about that.
3 Then we test under irradiation a statistically
4 significant quantity of fuel and with the monitoring
5 to know the in-pile performance and the PIE to
6 demonstrate that the requirements that we'll talk
7 about are actually met.

8 Do the same thing under accident
9 conditions and then use this data from the program to
10 either improve the models or to qualify the models.
11 So we have separate experiments to improve and then
12 the qualification data come from a completely
13 independent data set.

14 So these are the simple questions that
15 we'll answer. What are the reactors designer's
16 quality and performance requirements because then
17 I'll show you what I think, how we're doing relative
18 to that. Can the fabrication process meet those
19 requirements? And will the fuel be able to meet the
20 performance requirements under normal and accident
21 conditions?

22 How well do the models predict what's
23 being observed? You saw a little bit in Dave's talk.
24 I'll have a little bit about what the new fuel, what
25 we think is going on. And I'll also try to tell you

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1 a little bit about what we've learned. And we'll put
2 those answers to those questions in red in the
3 presentation so they jump out at you. Next.

4 So this is, you know, we don't actually
5 have a design that we're going to present at this
6 point. This is based on historical MHTGR designs
7 peak fuel temperature of 1400 C, a time average
8 maximum of 1250. The canonical 1600 C under
9 accidents that everyone knows about this fuel.
10 Burnup of 18 percent, fast fluence less than five,
11 10^{25} .

12 Now here's the quality specifications
13 that come from the reactor designer. These are the
14 major defect specifications. The contamination we
15 have missing the defective buffers, missing a
16 defective pyro carbon, defective silicon carbide,
17 missing or defective pyro carbon. And then the in
18 service failure rates under normal operation in core
19 heat-up accidents.

20 And I highlight the ones that are really
21 important that we can talk about today. The
22 contamination and the defective silicon carbide are
23 large drivers of the source term. Contamination is
24 a, you know, a uncontained uranium. So the fission
25 products from that would release.

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1 Defective silicon carbide, the cesium
2 will get through defective silicon carbide. Pyro
3 carbons will not be a good enough barrier. So you
4 tend to worry about those. And then those
5 incremental failures.

6 DR. KRESS: What kind of defects can you
7 have with silicon carbide? Is it the thickness of
8 the layer or the density of this?

9 MR. PETTI: It can be density, porosity
10 is probably --

11 DR. KRESS: Porosity.

12 MR. PETTI: -- and I will show you today
13 that is not what we worry about in the field today.
14 We're meeting, we're exceeding that specification by
15 a factor of three to five at 95 percent confidence.
16 I'm not at all worried about bad silicon carbide.
17 But I think this has to do and we'll talk about how
18 we make it. Technology today versus what the Germans
19 did is really good.

20 And then I want to talk about the, we'll
21 talk about the incremental failure rates 2 times 10^{-4}
22 and 6 times 10^{-4} and where we are. We think there's
23 a lot of margin there relative to the reactor. So
24 here is where we spent a lot of time, I showed the
25 process in a very simple overview. The top part is

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1 the laboratory scale process.

2 The bottom is the engineering scale. It
3 starts with making kernels in a Sol-Gel process. I
4 couldn't put all the pictures at B&W. They're our
5 vendor where they actually make the kernels. It's a
6 very involved process.

7 MEMBER CORRADINI: So B&W does both lab
8 and engineering?

9 MR. PETTI: No, Oak Ridge did the
10 laboratory --

11 MEMBER CORRADINI: That's what I thought.

12 MR. PETTI: -- and B&W did the industrial
13 scale. A lot of work on getting the UCO and making
14 really good UCO. And we've got that process. Then
15 you go to coating. And the laboratory scale, I call
16 it Coke can coating. It's a 60 gram charge, about
17 the size of your Coke can, is the active cylinder.

18 You wouldn't, you inject gases that
19 decompose, acetylene, propylene for the carbon
20 layers. You form a carbon on the particles. It's a
21 fluidized bed. For silicon carbide you use
22 methyltrichlorosilane, hydrogen and sometimes argon.
23 And in fact we're using argon as our base coating.

24 The industrial scale is bigger. It's
25 about a two kilogram charge in a six inch coater. By

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1 comparison the Germans used an eight inch coater,
2 five kilogram charge. That's what the Chinese still
3 use. There are sort of trade offs in processing
4 about what's the right size, you know. It depends on
5 how big the capacity is for the plant.

6 MEMBER ARMIJO: Just to get it straight
7 a coater puts on the silicon carbide?

8 MR. PETTI: The carbon layers and the
9 silicon carbide sequentially, each layer.

10 MEMBER ARMIJO: And what's the size of
11 the batch, is that the two kilogram?

12 MR. PETTI: Two kilograms, yes. That's
13 the two kilograms uranium, not even --

14 MEMBER ARMIJO: How much fuel does that
15 make?

16 MR. PETTI: In terms of number of
17 particles?

18 MEMBER ARMIJO: No, no, yes, why not.

19 MR. PETTI: Millions.

20 MR. SILADY: But relative to your
21 question, burning question, there is like eight or
22 nine billion in a reactor. So this makes a million,
23 this two kilograms.

24 MR. PETTI: So those, I'm not sure we're
25 going to get, yes, go back for a minute to answer his

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1 question. So these are the specs that you worry
2 about, do you have a good fuel. These are at the
3 batch level and there are specifications at a lot
4 level where you amalgamate batches.

5 MEMBER CORRADINI: And you sample to
6 determine --

7 MR. PETTI: Yes, so here to meet 2 times
8 10^{-5} heavy contamination at 95 percent confidence I
9 have to take a large amount of fuel.

10 MEMBER ARMIJO: And what do you do? What
11 do you do, dissolve it?

12 MR. PETTI: Okay, then you go through a
13 process called leach, burn, leach. You leach it in
14 acid and that gets the easily exposed uranium. Then
15 you burn all the carbon off, then you leach it again.
16 And so if there's a defect in the silicon carbide the
17 acid will go through and leach out. So you get the
18 contamination and the defective silicon carbide.

19 MEMBER ARMIJO: Okay. So that's
20 destructive and characterizes a batch.

21 MR. PETTI: Yes.

22 MEMBER ARMIJO: What about the actual
23 silicon carbide integrity? Is that also with that
24 leach, burn, leach?

25 MR. PETTI: Yes, if you had bad silicon

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1 carbide the acid would get in and you'd know it. You
2 also, we have specifications on all the thicknesses
3 all the densities.

4 MEMBER CORRADINI: But that requires an
5 inspection of a different sort then --

6 MR. PETTI: Yes, you basically take some
7 particles and you section them and you have a
8 computer that will calculate the thicknesses. That's
9 what we do there. And then we also have anisotropy
10 specifications on the carbon layers. I think those
11 are the, all of the major.

12 MEMBER ARMIJO: Okay. So there's no
13 nondestructive?

14 MR. PETTI: We continue to look and try
15 to develop those. But the more you look, you know,
16 you go back to this because you know it works. It's
17 really hard. We have not been able to develop
18 something that is, unfortunately. A lot of the
19 effort in the last two years for us has been making
20 compacts.

21 MEMBER CORRADINI: So can I ask another
22 site question, Dave? So have you asked the staff to
23 comment on your fuel sampling to meet those specs or
24 is that yet to be done by whomever chooses to be the
25 owner, operator of this thing and order the fuel?

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1 MR. PETTI: I believe our White Paper
2 talked about that approach and we had several RAIs on
3 it.

4 MEMBER CORRADINI: But staff as part of
5 their current activities was not to review and
6 comment specifically on this.

7 MR. PETTI: No, I think they did.

8 MR. KINSEY: I guess as a point of
9 clarification as Dave said, we covered that topic in
10 our White Paper. The staff asked us some questions
11 through some RAIs that we responded to. You know and
12 if there was a concern about the process we're using
13 for sampling we would expect that they would tell us
14 that as part of our overarching question over are we
15 missing anything.

16 MEMBER CORRADINI: Okay, thank you.

17 MEMBER ARMIJO: Just to make sure each
18 batch is tested to meet the spec, that's what you
19 said?

20 MR. PETTI: Yes. So go back, to make the
21 compacts is really challenging. You only need 400
22 thousand pebbles, 450 thousand pebbles in a pebble
23 bed of 600 megawatts. You need like three, six
24 million one inch compacts. So the throughput is much
25 different. So we started with the German process for

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1 overcoating and pressing. And they use like a Betty
2 Crocker mixer to overcoat.

3 And it just was a multi-step process that
4 when you looked at it from our throughput just wasn't
5 going to work. So we had to develop a really
6 completely different approach. And in fact it's just
7 been very, very successful. You have to make this
8 matrix and that's a complicated, it's a graphite
9 flour and you put in the resin and you mix it
10 together.

11 And we decided to go with a dry jet-
12 milled product that the fuel vendor could buy from a
13 supplier so that he didn't have to have large
14 amounts. It's a carbon dust basically which you
15 don't really don't want to necessarily have to do
16 yourself. And very uniform which is really
17 important, I think, in the overcoating.

18 Then we went and we bought a overcoater
19 that the pharmaceutical industry uses to overcoat
20 pills and the contact. That picture there, it's a
21 large armoire size, is their lab scale, which just
22 talks about our medical industry in the United
23 States. They have much bigger ones. That's a
24 production unit for what we're going to need.

25 MEMBER ARMIJO: Is there a binder or

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1 anything or is that just dry?

2 MR. PETTI: Yes, it's a binder. Water,
3 we that's another interesting story. The Germans use
4 methanol. We didn't want to use methanol because
5 it's flammable in a fuel facility. We tried water
6 and it worked. I mean it stuck, it held it together.
7 And you'll find that the overcoater, so you put the
8 fuel in, put the matrix in, you put the water in 100
9 percent yield and the particles are better than what
10 you get.

11 MEMBER ARMIJO: So you squish them and
12 dry them and away you go?

13 MR. PETTI: Yes. It's great. Then you
14 press them. We have an automatic presser. We can do
15 about four compacts in 90 seconds. So you just,
16 you'd triplicate this, automatic feeding. And then
17 there's some heat treatment. So really nice. So we
18 now have a full pilot line.

19 MEMBER ARMIJO: And this is at B&W?

20 MR. PETTI: This is at B&W. So, next.
21 So we have basically reestablished the capability to
22 make this fuel since last time the MHTGR was around.
23 A lot of effort in understanding how to fabricate it
24 which we think is really important. That it really
25 isn't an art, that it is a reproducible. There is

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1 some science and engineering.

2 And today we're fabricating high quality,
3 low defect fuel. We can meet the physical
4 specifications and we're almost meeting the design of
5 the defect specifications. The heavy metal
6 contamination is at two 10^{-5} , sometimes we're at 2.5,
7 sometimes we're at 3, sometimes we're at 1.8. We're
8 in that mode trying to ring that out, particularly at
9 95 percent confidence.

10 We think with larger sample sizes and as
11 we continue to mature the process we should be able
12 to meet the defect specifications in a true
13 production mode. The Chinese have done it. The
14 Japanese have done it. We certainly don't think it's
15 a problem. We have a vastly improved quality
16 reproducibility and process control and
17 characterization of the fuel.

18 One of the things that was important was
19 control of the process. We use mass flow controllers
20 with the gases and that gives you really nice, tight
21 control that didn't exist when the Germans did it.
22 We removed every high-variability human interaction
23 in the process. We do not table these particles like
24 the Germans did. We thought that we, we did a lot of
25 work and found we were throwing away good material.

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1 And we went with precision sieving which is used in
2 many, you know, industrial settings.

3 So we've tried to bring today's
4 technology to bear. And you see it. I mean the
5 other one is making the silicon carbide. We're using
6 an evaporator from the chip industry. You know,
7 they've spent billions probably to make chips. You
8 see it. We just see a better, you look at the cross
9 section, the micrographs, it just looks better. You
10 can tell it's really good. And it just has to do
11 with where technology is today.

12 So we think establishing this vendor and
13 the associated understanding really lends some
14 credibility that what the Germans did in the 80's is
15 repeatable and has a sound basis. And so all the
16 technologies for this pilot line are in industrial
17 hands and we'll be making the final qualification
18 fuel in 2013.

19 So let's turn now to performance. This
20 is, I've shown this before our radar plot for the
21 five key parameters. Just a note that the brown
22 curve is what we're trying to do. The dark green is
23 the Germans and the light blue is the Japanese. So
24 we do have a more aggressive performance envelope
25 then historically done. But we have in fact been

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1 testing at these conditions and then having had
2 success.

3 The packing fraction is higher than the
4 last time I showed you because we've moved, based on
5 design recommendations moved to a higher packing
6 fraction.

7 MEMBER CORRADINI: And the packing
8 fraction you define as the ratio --

9 MR. PETTI: Particles to the binder, to
10 the compact volume, yes. So this is the program in
11 one slide. Eight experiment campaigns AGR-1 through
12 AGR-8. They each have a different purpose. AGR-1
13 was the laboratory scale fuel. AGR-2 is what we call
14 performance demonstration. You could call it a dress
15 rehearsal prior to the official qualification. It is
16 a large coater, industrial scale fuel.

17 AGR-3&4 is got fuel that will fail in
18 reactor to deal with addressing source term issues.
19 We'll talk about that. Five, six is the
20 qualification tests. Seven and eight are for
21 validating fuel performance codes and fuel margin
22 testing and then fission product behavior validation.

23 And then beyond the irradiation there's
24 a parallel campaign with that irradiated material to
25 do safety testing and PIE. And then moisture and air

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1 ingress effects are part of AGR-5&6. We plan to
2 develop furnaces that we can put air and moisture
3 quantities in and test the performance of the fuel,
4 the fuel in the graphite body irradiate a graphite
5 body even to do those sorts of tests.

6 Without doing anything in pile with large
7 amounts of moisture. It's been done in the past and
8 it's just not a big deal. Moisture is not really a
9 problem with the fuel. So in terms of AGR-1, a
10 slightly different particle.

11 We started with 350 micron UCO TRISO, 19
12 percent enriched. Goal burnups were 18 to 19
13 percent. We exceeded that a little bit. We went to
14 about 19.5, 19.7. Peak time average temperature less
15 than 1250. Average, volume average temperature may
16 be around 1150. It took almost three years to do the
17 irradiation and we had a very healthy population of
18 particles, 300 thousand reached burnup of 19 percent
19 with no failures.

20 They were tested in six individual
21 capsules shown there and they each were individually
22 controlled on temperature and that control gas is
23 swept out into fission product monitors so if a
24 particle fails you see a gas release, I mean we know
25 if that's the case. So it took the Germans 15 years

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1 to accumulate the statistics to get that many because
2 they were testing one, two, maybe three pebbles at a
3 time. The volume of testing that we have in the ATR
4 allows us to test a lot of particles very quickly,
5 which is really good.

6 Okay, the next is a plot of the
7 temperatures, the temperature census in the each of
8 the six capsules shown on the right. And then
9 temperature distribution expected, this is the
10 conceptual design that GA did for the NGNP called the
11 SC-MHR. And if you just look at it where it's, the
12 experiment is a much more conservative in terms of
13 the average temperatures of the capsules were
14 somewhere between a 1000 and 1100. The average
15 temperature in the core sits around 725, it looks
16 like.

17 And then the peak temperatures, we've got
18 a very large amount of fuel out at peak temperature
19 much greater than you could expect in an NGNP. Next.
20 This is another slide that not everyone has seen.
21 This is relatively new so Don, you should be looking
22 at this. We've taken the temperature predictions
23 for, we basically have a finite element model and
24 every finite element is about a particle. It tells
25 you how detailed it is.

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1 And we've basically created a time at
2 temperature census. So this plots how much of the
3 fuel was at what temperature for how much time. So
4 take a look at the purple at 1300 degrees you can see
5 that at a hundred hours about ten percent of the fuel
6 saw temperatures in excess of 1300 degrees for a
7 hundred days.

8 DR. KRESS: Each one of those dots
9 represent a kernel?

10 MR. PETTI: Yes, basically each represent
11 a particle in the test.

12 MEMBER CORRADINI: And the line
13 represents what?

14 MR. PETTI: The line is sort of the
15 average of those colors, the software will put like
16 a --

17 MEMBER CORRADINI: So it's the average of
18 the population at a time.

19 MR. PETTI: Yes, it's the average of that
20 color, that strip. So you can see in the lower
21 corner the little red. We saw five percent of the
22 fuel, maybe three percent of the fuel greater than
23 1400 for 50 days. So this fuel saw a lot of time at
24 high temperature. And so --

25 CHAIR BLEY: And irradiation at the same

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1 time?

2 MR. PETTI: Yes under irradiation. And
3 that's why I think some of the, we're seeing a lot of
4 silver release. This is really because of the time
5 and temperature.

6 So in terms of AGR-2, again this is
7 vendor produced both UO2 and UCO at the time. We
8 decided to, we had such success with AGR-1 that we
9 would look at some pebble type fuel, 500 micron UO2.
10 This South African fuel produced by South Africa,
11 fuel produced by the CEA in France, part of our Gen
12 4 collaboration. We made UO2 at B&W. They can make
13 either. Not a problem. It's one, you just don't put
14 the carbon in and it's pretty much the same process,
15 change centering schedule.

16 DR. KRESS: Going back to your previous
17 slide, you don't have to go to, why does the
18 temperature decrease, is it because you're using up
19 the uranium?

20 MR. PETTI: We're holding it constant.
21 But why would there, yes, this has to do with the
22 detailed operation.

23 DR. KRESS: How you operate the system?

24 MR. PETTI: Yes, at the very end of the
25 experiment when the temperatures are dropping because

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1 there is no uranium, we're trying to keep them up by
2 changing the gas mix. And we have some controls in
3 ATR to change the power, move some reflector and we
4 kind of overshoot. We kept saying we've got to keep
5 the temperature up and we ended up running it
6 actually very hot at the end so.

7 So notice the UO2 425 micron UCO, two
8 capsules at 1250, one at 1400. We're calling it an
9 early margin test. This is really an AGR-7 objective
10 but we had the space. With the recommendation made
11 from GA at the time and we thought it was a good one.
12 And then the UO2 is much more pebble bed, 9.6 percent
13 enriched, 11 percent FIMA. The French enrichment is,
14 that what's they had so that's what we tested.

15 So it's a mixture. But it's really nice
16 in this capsule each one's a different conditions.
17 But we can do this.

18 DR. KRESS: Is the pebble bed then going
19 to have this migration of the kernel problem?

20 MR. PETTI: The gradients in the pebble
21 bed probably not as much, yes.

22 MEMBER ARMIJO: David, all these
23 different fuels that you've tested, did you test them
24 in the form of compacts or as particles?

25 MR. PETTI: Compacts.

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1 MEMBER ARMIJO: Compacts. So they sent
2 you compacts or --

3 MR. PETTI: The French sent us compacts.
4 The South Africans sent us particles and we compacted
5 them. So it's a particle test more than a --

6 MEMBER CORRADINI: Pardon my, that I
7 don't know the unit or I forgot it. Remind what FIMA
8 is.

9 MR. PETTI: Fissions per initial metal
10 atom. Think atom percent.

11 MEMBER CORRADINI: And the enrichment is
12 weight you're saying?

13 MR. PETTI: Yes, yes.

14 MEMBER CORRADINI: But what I want to
15 understand is when the number is at or below the
16 enrichment, I'm okay. When it's higher does that
17 mean I'm doing some transmutation and burning?

18 MR. PETTI: Well, yes, you're doing that
19 anyways even if they're lower ones, but, yes, yes.

20 MEMBER CORRADINI: Okay. But one is atom
21 percent and weight percent to process it.

22 MR. PETTI: So this is a plot of the gas
23 release the R/B. You heard that earlier. Think of
24 it as a release fraction. I thought all the old US
25 experiments post Fort St. Vrain there and all the

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1 German experiments on the right. The blue crosses
2 are the six capsules for AGR-1. So clearly as good
3 as the German fuel at twice the burnup, German fuel
4 would go to about nine to ten percent FIMA.

5 And the little box is where we are today
6 on AGR-2. It's higher because there is an exposed
7 kernel, a defect in every, again at industrial scale
8 in almost every capsule. And then the hot capsule,
9 the gas release from a hotter particle you get a
10 little bit more gas release.

11 MEMBER CORRADINI: Did you tell us what
12 you did to make a defect or that's coming? I forgot.
13 I know you've told us in the past. I just forgot.

14 MR. PETTI: We believe that upon
15 unloading of our coater we're damaging particles.
16 And that's what the defect is. It's not inherent in
17 the process.

18 MEMBER CORRADINI: Well I guess I'm
19 asking about the pink. Are you answering the blue?

20 MR. PETTI: No, I'm answering the pink.

21 MEMBER CORRADINI: And so you consciously
22 did that?

23 MR. PETTI: No, no. That was
24 inadvertent. Inadvertently, how you take them out of
25 the, this is a, it's an issue, it's in the handling

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1 between the coating and the compacting. And we're in
2 an HEU facility. And I just can't tell the guys to
3 go change out this thing because they live under very
4 strict rules so everything's treated as HEU even
5 though it isn't. It would take me a year to take a
6 valve out of the system because it's part of their,
7 a permanent part of their system.

8 MEMBER CORRADINI: So you because they're
9 treating it as if it's HEU you think they're damaging
10 it?

11 MR. PETTI: Well just the physical
12 configuration. Some of the stuff that's there, so I
13 said we'll get rid of that. And they said no, we
14 can't. I mean it just, so in a real process line you
15 could design from scratch you won't have that issue,
16 you know.

17 MEMBER CORRADINI: It's avoidable.

18 MR. PETTI: It's avoidable, clearly an
19 avoidable. So we're trying our best to work around
20 it. That's basically, so --

21 MEMBER ARMIJO: But it just shows how
22 sensitive it is to --

23 MR. PETTI: Yes, yes. Once you get them
24 compacted they're great. But you've got to get them
25 compacted. Now let me turn to the source term. I

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1 think you've heard this. We're going to have a
2 mechanistic source term taking into account all the
3 different values. The goal of our program is to
4 provide the technical basis to support the design and
5 licensing. And there are three experiments for two
6 major campaigns called AGR-3/4 and AGR-8 and the
7 follow on PIE safety test and loop testing. And AGR-
8 8 is the independent validation part of the plan.

9 DR. KRESS: Will we see the details of
10 these fission product models, eventually?

11 MR. PETTI: Not, maybe eventually, not
12 today. But --

13 DR. KRESS: I mean I knew not today.

14 MEMBER CORRADINI: They gave us a hint at
15 it in past presentations. You were there.

16 MR. PETTI: And I think in the White
17 Paper there's some discussion in the appendixes
18 maybe. I'm trying to remember.

19 MR. ALBERSTEIN: There's an appendix with
20 a fair amount of detail on transport mechanisms.

21 MR. PETTI: A little bit on the RAIs.

22 MR. ALBERSTEIN: And later in Dave's
23 presentation he's going to talk about a couple of
24 these aspects.

25 MR. KINSEY: Excuse me, Dave, before we

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1 move on. This morning we talked a little bit about
2 the inside-out look versus the outside-in look. I
3 was going to maybe ask either Fred or Dave to spend
4 a minute on that and make sure that what we were
5 communicating was clear as we're going through this
6 if that's all right?

7 MR. SILADY: Yes, I'll be happy to just
8 add some footnotes here maybe. I'm not a LWR guy by
9 any means. I'm just a CGR. So I don't know
10 comparisons. And oftentimes it's best to stay away
11 from comparisons. But you know our barriers and we
12 know that the fuel is the most important and it's
13 receiving the most emphasis. And the silicon carbide
14 is the most important barrier.

15 And we work inside out in that sense. We
16 put more, tighter requirements on that fuel as
17 opposed to helium pressure boundary or certainly to
18 the reactor building. My understanding on the
19 existing reactor barriers they have a clad, they have
20 a pressure boundary and they have a containment.

21 And certainly if they have a problem with
22 the clad they still have the helium pressure boundary
23 there and they still have the containment there. But
24 if they have a problem with the helium, not the
25 helium pressure boundary, but the reactor coolant

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1 boundary their clad is linked to that in the sense
2 that you have to keep the core covered or the clad
3 goes. And then you've got much greater release that
4 the outer barrier, because it's passed the coolant
5 boundary's barrier, the outer barrier has to do
6 yeomen's work to meet the requirements.

7 In our case, if we have a problem with
8 the helium pressure boundary, it's okay. We still
9 have the radionuclides in the fuel. There's no
10 linkage. We don't have to keep the core covered with
11 helium. It will operate at pressurized or
12 depressurized, circulated and if it has any pressure
13 it will be natural convection otherwise we'll heat up
14 and we'll cool down. And the linkage isn't there.

15 And we only see a small fraction, a small
16 increase coming from either the initial or the
17 delayed release that goes into the reactor building.
18 And so it's been designed to do the functions it
19 needs to do, which is more focused on structurally
20 protecting from external vents rather than being a
21 radionuclide barrier.

22 So this concern about well you only have
23 one thing left and it's not as good because it's a
24 vented building and it's not a containment. I don't
25 think that analogy is the right way to think about it

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1 because we don't have that ability or our silicon
2 carbide to fail when we lose helium out the helium
3 pressure boundary. So I just wanted --

4 MEMBER ARMIJO: In case you got the
5 impression that's what I was saying, that's not what
6 I saying. You just basically have a system that
7 doesn't need three independent barriers because of
8 the nature of your overall system. I don't have a
9 problem with that. Just, but there aren't three,
10 there's only two. There's the silicon carbide and
11 your pressure vessel which are the only, what I call
12 physical barriers.

13 MR. SILADY: I think we have more
14 independence in our barriers then --

15 MEMBER ARMIJO: I think you have great
16 fuels so don't, so let's not argue about it.

17 MR. SILADY: Good. Thanks. I feel
18 better.

19 MR. PETTI: So let me turn back now. So
20 this is the first source term experiment in the
21 program AGR-3/4 to understand the behavior of the
22 fission products from that small fraction of defected
23 fuel. How much retention is the graphitic components
24 in the core? And we use something called designed-
25 to-fail fuel. So this is fuel, so we have a known

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1 source of fission products. So these are particles
2 with no silicon carbide and very bad pyro carbon that
3 we know will fail under irradiation.

4 DR. KRESS: That's how you designed it?

5 MR. PETTI: That's how you design the
6 kernel.

7 MEMBER CORRADINI: You cooked them to be
8 that way.

9 MR. PETTI: Right. We made them --

10 MEMBER ARMIJO: You fabricated them
11 without silicon carbide.

12 MR. PETTI: Right and very anisotropic
13 pyro carbon. So it rips itself apart under
14 irradiation.

15 MEMBER ARMIJO: Okay. So this is about
16 as bad as you can get?

17 MR. PETTI: Right. And then we, very
18 carefully Oak Ridge developed a technique to put them
19 all in the center of the compact. It was really
20 cool. So we know that right, so we know that
21 temperature. So we know, you know, really well. And
22 then they even X-ray radiographed them so we know
23 they're all on the center. And they all failed over
24 Christmas. They always, it always happens over
25 Christmas. I get the phone call, you know.

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1 So and the goal here is to establish the
2 transport of the metallic fission products and the
3 retention in the graphitic components and the release
4 from exposed kernels as a function of burnup,
5 temperature and fluents. So we have, there you see
6 the capsule. The inner ring is the fuel. The
7 tannish striped one is an annulus cylinder of matrix
8 material.

9 Then the outer one, the silvery grey is
10 the fuel element graphite, the block graphite. And
11 then the outer graphite with the holes is a sink that
12 is there to, so no fission products go beyond and
13 also for our instrumentation to go in those through
14 tubes.

15 MEMBER CORRADINI: So the thought is the
16 junk leaks out of the center hole to the outer ring?

17 MR. PETTI: Right. One dimensional
18 diffusion as best as possible.

19 MEMBER CORRADINI: And it is captured
20 there?

21 MR. PETTI: Right. Now we're monitoring
22 for fission gas. So we're getting a release as a
23 function of time which can, you can call to burnup
24 and fluents. And there are 12 of these capsules
25 stacked. So AGR-1 had six. This has 12 in the,

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1 what's called the northeast flux trap in ATR, a
2 different location. And so we're able to get a
3 really broad range of temperatures and burnups. And
4 the matrix is quite complicated to show. But it kind
5 of, we're trying to envelope the core that we'll get
6 different combinations and be able to establish the
7 functionality.

8 MEMBER CORRADINI: And these are shorter
9 irradiations because you know as you cook them to
10 temperature and fluents they're going to fail early
11 on?

12 MR. PETTI: These failed in the first
13 week as expected. Yes.

14 MEMBER CORRADINI: And then you're going
15 to hold them there --

16 MR. PETTI: And then you hold them at
17 that temperature.

18 MEMBER CORRADINI: For not again, not for
19 three years though?

20 MR. PETTI: No, for about, it's about 400
21 to 450 full power days, which may be 18 months to two
22 years in the reactor.

23 MEMBER ARMIJO: Dave, did you ever
24 deliberately do any experiments in which you took the
25 silicon carbide, didn't make it to full thickness

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1 just, you know, stopped coating it at a certain point
2 and then test that fuel to see how long it would
3 perform just to get a feel for the margin that you
4 have in the retention?

5 MR. PETTI: We've never made, but for
6 instance the Japanese silicon carbide is only 25
7 microns instead of 35 microns, showing very, very
8 good behavior.

9 MEMBER ARMIJO: Yes, I would guess, the
10 reason I'm getting to that point is that in a batch
11 process when you do the qualification you, at least
12 in light water reactor fuel, you go off-normal, you
13 test your process controls and say well the lowest
14 temperature I could ever be at would be somewhere in
15 here. The highest I'll ever be within my process
16 control range. Did you, is that part of your fuel
17 qualification that, to check how sensitive you are to
18 slightly off-normal in your --

19 MR. PETTI: We have not done it through
20 testing. We've done some of that through analysis
21 right now. And we see no change in predicted failure
22 probability of the fuel over the specification range.
23 You have to get down below say 20 microns. But our
24 spec, we would not allow, we're allowing like 35 plus
25 or minus three microns.

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1 MEMBER ARMIJO: But so you're saying we
2 didn't do the process qualification that way.

3 MR. PETTI: The problem is there are so
4 many parameters. It gets so big.

5 MEMBER ARMIJO: So okay. I understand
6 how you detect the defects in the silicon carbide,
7 the leach, burn, leach. How do you detect either
8 non-uniform or very thin silicon carbide? Somehow
9 your process didn't --

10 MR. PETTI: We measure, we take a bunch
11 of particles and we slice them and we measure the
12 thicknesses and we make, you know, so we get a
13 distribution.

14 MEMBER ARMIJO: Okay. It's by optical --

15 MR. PETTI: By optical metallography.

16 MEMBER ARMIJO: Got it.

17 MR. PETTI: And we also measure
18 sphericity, that's the other thing I didn't say. And
19 these are very, these are much more spherical
20 particles than others use. So we think that's also
21 very beneficial. So in terms of the accomplishments
22 --

23 CHAIR BLEY: Is that different than the
24 old pictures we saw of these that to me are rather
25 irregular shaped particles?

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1 MR. PETTI: Well partly you get faked out
2 because of the magnification.

3 CHAIR BLEY: Of course.

4 MR. PETTI: But if you compare our fuel
5 to the German fuel we are more spherical. It has to
6 do with how we make the kernel actually compared to
7 the rest of the world. We use something called an
8 internal gelation process and they use external. And
9 this goes back 25 years where the US stood behind
10 that process and the rest of the world said no, we're
11 going to go the other way. And now they're
12 scratching their head when they represent stuff and
13 they ask the question remember that argument?

14 We're seeing the results. So we have
15 completed the most successful irradiation of fuel in
16 the US to 19.4 percent FIMA. We have confirmed the
17 expected superior radiation performance in UCO at
18 high burnup. We have see in the PIE no kernel
19 migration, no evidence of carbon monoxide attack of
20 silicon carbide, no indication of silicon carbide
21 attacked by the lanthanides.

22 Now if I take the fact that we saw zero
23 failures out of 300 thousand and I do the 95 percent
24 confidence estimate, that's below 10^{-5} . That is a
25 factor of 20 better than the reactor requirement of

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1 two 10^{-4} . That plus the fact that this irradiation
2 was much more severe in terms of temperature says
3 that I think there's substantial margin.

4 AGR-2 is underway. No failures to date.
5 Should complete by the end of this fiscal year.

6 MEMBER CORRADINI: Then I'm confused
7 about your pink. What am I missing about the pink
8 which I thought was --

9 MR. PETTI: The pink is gas release.
10 Okay. This is failure effects. You have to take the
11 release per failed particle, that's what collates the
12 two.

13 MEMBER CORRADINI: Say it again slowly.
14 I'm sorry.

15 MR. PETTI: That is gas release. This is
16 failure fraction. And so when a particles fails
17 there is a failure release fraction per particle
18 that's a function of temperature that collates the
19 two. And the number of particles in the --

20 MEMBER CORRADINI: So can I say it
21 differently. So not to go back to the curve, but if
22 I look at slide 20 the blue x's translates into a
23 failure fraction of --

24 MR. PETTI: Less than 10^{-5} .

25 MEMBER CORRADINI: -- less than 10^5 . And

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1 the pink ought to translate into something less than
2 10^{-4} ?

3 MR. PETTI: Yes. So we think that
4 there's margin here. In terms of AGR-3/4, it's
5 underway. All the particles have failed. We're now
6 kind of in a steady state mode. It started at this
7 last Christmas. So we've go initial failure and then
8 it kind of levels out. And so this data will be
9 absolutely critical for us for the source term
10 evaluations.

11 Now let me turn to PIE, which is the
12 other sort of big new thing probably since the last
13 time I talked. It took a lot of effort to get the
14 infrastructure in place to do this work. PIE on this
15 fuel is a lot different then pellet clad fuel given
16 the size of the particles and the like.

17 We have three major objectives. A
18 detailed characterization of the fuel after the
19 irradiation in the reactor. A mass balance of the
20 fission products for the source term so that you know
21 that something is, has or has not been released. And
22 then the high temperature safety testing to establish
23 the fuel behavior under accident conditions.

24 And I show the two furnaces in Idaho and
25 Oak Ridge that are used to do the safety testing.

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1 And then some of the detailed techniques to handle
2 these particles. We can deconsolidate the matrix and
3 then retrieve the irradiated particles and do things
4 with them at the particle level. It's an
5 electrochemical process in an acid. So you slowly
6 basically dissolve the matrix. It's not, it just.

7 MALE PARTICIPANT: I didn't know you
8 could dissolve graphite.

9 MR. PETTI: Not dissolve, but that's not
10 the right word. You can see in the picture there's
11 shards, there's pieces of graphite. You just
12 basically, it comes apart. And so we handle it like
13 the eye doctor does the eye surgery. We have a big
14 tv screen and plus on that in a hot cell. It's
15 pretty amazing.

16 So the other thing that we're doing
17 that's new is that we are basically throwing every
18 technique that there is in material science at the
19 fuel. The lab has received a number of real state of
20 the art instruments to look at the nanoscale to do on
21 irradiated material. And so the far right is a
22 picture of a FIB a TEM sample of the silicon carbide
23 layer.

24 We do the typical work with the SEM at
25 the micron level. We also look above the

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1 micrographs. So we're getting the whole scale to
2 look at everything. That black funny shaped object,
3 that's a precipitate of palladium probably. So we're
4 really looking at --

5 MEMBER CORRADINI: Where are you talking
6 --

7 MR. PETTI: Nanometer, the nanometer.
8 That's that right there. So we're looking at a
9 different level than has ever been looked at the
10 fuel. It's very exciting. We don't have answers for
11 everything yet. But that's what we're doing to
12 really try to understand what's going on in the fuel.
13 So we expect that we're going to learn an awful lot
14 more.

15 MEMBER ARMIJO: Are you studying the
16 silicon carbide characteristics changes as a, with
17 this technique?

18 MR. PETTI: Yes, we can do some work
19 looking at irradiation effects as well. The other
20 big thing that we've done is we've developed a
21 methodology that if there is a defect that is
22 contributing to the release that we're able to find
23 it. We kind of call it the needle in the haystack.

24 So we start with 300 thousand particles
25 in the capsule. We gamma scan the graphite that hold

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1 the compacts. There you see the picture, the three
2 stacks. You look for hot spots. You do a tomography
3 and there's a couple hot spots. Let's take that fuel
4 and deconsolidate it and get the particles.

5 So actually that's where it goes from
6 Idaho to Oak Ridge because they can look at every
7 particle. We can't look at every particle, we can
8 only look at only 60 to 100. They can look at all of
9 them in the compact. And we put them in a machine
10 called IMGGA, which is basically a gamma spec. So
11 every particle gets a gamma spec and you find them
12 all with low cesium because that's what you saw, you
13 saw some release.

14 Take that, put it in an X-ray
15 thermograph. There's a 3-D reconstruction and we can
16 find the defect and the nature of the defect. And in
17 this case we know that this was a defect caused at
18 laboratory scale that we're pretty sure that doesn't
19 occur at industrial scale. But it gives us a heads
20 up to okay, we better look at this on AGR-2 as well.

21 So never have we been able to go this
22 level before. You'd see a release and you'd scratch
23 your head after a furnace test and not understand it.
24 We now today can go and find the particle that's
25 defective. It takes a lot of effort. But we think,

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1 a) it helps explain it, it takes the mystery out of
2 why there was a release and just helps with the
3 understanding. So this is a long process.

4 But we have basically found, we knew how
5 many defective particles based on the quality control
6 data we would expect to see. We have found them and
7 we have characterized them.

8 MR. ALBERSTEIN: This is not to be
9 confused with the earlier discussion about locating
10 failed fuel in an operating reactor. That's a whole
11 different --

12 MEMBER ARMIJO: That's a different thing.

13 MR. PETTI: It's hard enough to do --

14 (Crosstalk)

15 MEMBER ARMIJO: You're actually seeing
16 the cracks in the silicon carbide?

17 MR. PETTI: Yes.

18 MEMBER ARMIJO: And on an individual
19 particle, particles?

20 MR. PETTI: And that is a bad --

21 MEMBER ARMIJO: Might get some idea of
22 what might have been the reason.

23 MR. PETTI: What caused it, right.

24 MEMBER ARMIJO: Compare a non spherical,
25 at least those two.

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1 MR. PETTI: Yes, well that's swelling.
2 This is after irradiation and there's a crack in the
3 buffer and so the kernel actually extrudes into the
4 buffer.

5 MEMBER ARMIJO: It does.

6 MR. PETTI: Yes, from swelling.

7 CHAIR BLEY: Well plus look on here.
8 You've got, is that the same?

9 MR. PETTI: Yes, it's the same one, yes.

10 CHAIR BLEY: Yes, because up here they've
11 got it elongated this way.

12 MR. PETTI: They basically let through
13 the cracks. The other thing this is, you know, you
14 can't show it here. But this is now 3-D, which is
15 also something that we've not been able to do. When
16 you section, you know, you're taking your chances
17 what do you find? 3-D we've got a completely new
18 look at things which is very exciting to understand.

19 We know that this was caused from poor
20 fluidization in the laboratory scale coater. The
21 particle hit the wall and then there's soot on the
22 wall and it causes an interruption in the silicon
23 carbide layer. And if they're small it's not a
24 problem. This one happened to be fairly large and it
25 does cause a problem.

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1 MEMBER ARMIJO: Very nice.

2 MR. PETTI: So there are a couple things
3 now that we've learned from the PIE that are follow
4 up items that the NRC has put in their assessment
5 reports. And we wanted to give you a sense of what
6 that is. It's, the questions probably two pages and
7 the answers probably five pages. But I'm going to
8 try to get you sort of snap shot.

9 First is that the irradiations in ATR
10 don't produce enough plutonium. It has to do with
11 the spectrum and the fact that we borated the
12 capsules so we're absorbing some of the thermal
13 neutrons in the boron. And then we don't get enough
14 palladium because you get a lot more palladium from
15 plutonium fission than you do uranium fission.

16 And we worry about palladium corrosion of
17 the silicon carbide at high burnup. So the question
18 is well how do you know that the test is
19 representative? So we did a lot of calculations.
20 And we found that what we get in AGR-1, about 40
21 percent below, for silver 40 percent below that
22 expected in the reactor and palladium about 33
23 percent below that expected in the reactor at the
24 peak burnup.

25 If you go to more average burnups the

1 numbers are smaller. So we went back and looked and
2 the point for me that was the most convincing was
3 that there was an experiment done back in the old
4 days of pure plutonium fuel. So all fissions are in
5 plutonium in Peach Bottom in a gas reactor for about
6 a thousand days called FTE-13. Went to 70 percent
7 FIMA. Typical temperatures and levels of damage.

8 Some palladium interaction was observed
9 but no large scale degradation of the silicon carbide
10 layer was observed by palladium. When you look at
11 the volumetric concentration of palladium in those
12 kernels it's about 75 times that in AGR-1. And the
13 surface, if you say it's not volume it's surface,
14 you'll go surface concentration, let's say around the
15 kernel surface of a silicon carbide surface, it's
16 60x.

17 So we felt that the effect, the small 33
18 and 40 percent effects were small although the fact
19 that there are tests out there with a lot more
20 palladium. So we're continuing to look for palladium
21 because the historical data suggests that this is
22 something that should be a concern.

23 MEMBER ARMIJO: What's the mechanism by
24 which palladium damages silicon carbide?

25 MR. PETTI: I think if it's in high

1 enough concentrations there are silicides on the base
2 diagrams, eutectics low melting point.

3 (Crosstalk)

4 MR. PETTI: The palladium, yes. The
5 palladium silicide. And there's a number of them on
6 the phase diagram. But to date we have not seen any
7 palladium attacked in AGR-1, which is actually
8 amazing when you think about the times and the
9 temperatures that we were operating at.

10 MEMBER CORRADINI: So you feel these
11 others have bounded what you would expect to see as
12 an effect?

13 MR. PETTI: I think that, yes. I think
14 it's a couple things. I think accelerated
15 irradiations, our historical database may be wrong
16 because if they're all based on highly accelerated
17 irradiations. Also it gets into the detail of the
18 interface between the pyro carbon and the silicon
19 carbide layers.

20 We have a different sort of interface
21 then the historic US fuel. It's more German-like.
22 And the Germans never saw this effect big. This was
23 bigger in American fuel. So it's a combination of I
24 think the, how we make the fuel and the fact that we
25 used to test under very accelerated conditions.

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1 MEMBER CORRADINI: So to ask, so this is
2 your response to the RAI. And have you heard back
3 from the staff on this relatively?

4 MR. PETTI: I think this remains a
5 follow-up item. We just disagree. And we're going
6 to, it's not like we're not going to keep looking.
7 Of course we're going to keep looking in the PIE.

8 MEMBER CORRADINI: I understand. I just
9 want to understand.

10 MR. PETTI: Right. But you'll hear from
11 them I'm sure on it. So we wanted to give you sort
12 of our perspective. The second one follow-up was
13 they asked some questions on whether silver or
14 palladium release, if you had a lot of it, would you
15 degrade the silicon carbide? And there's some old
16 theories, some old publications from the Germans that
17 thought that, you know, if you could degrade the
18 silicon carbide the cesium would come out.

19 And understandably you're worried about
20 cesium. And then is there some sort of an
21 enhancement under irradiation for cesium? So simply
22 there's no evidence of this in the German database.
23 Now under AGR-1 we released a lot of silver, 30, 40,
24 50, 60 percent in some compacts. One percent of the
25 palladium is outside the silicon carbide in AGR-1.

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1 So there are lots of these fission products have gone
2 through.

3 Cesium is low. There is no release of
4 cesium. If you see cesium, it means there's a
5 defective particle because when we do the measurement
6 we'll see the uranium as well. So the silicon
7 carbide seems to be very, very good and not
8 susceptible to this theory of silver and palladium
9 degrading the silicon carbide, we just don't see any
10 data yet.

11 So we're seeing minimal release of cesium
12 in the matrix. And if there were palladium
13 degradation you should see a lot more cesium. I mean
14 we're not seeing anything, not even one particle's
15 worth.

16 MEMBER CORRADINI: So can I ask you a
17 different question?

18 MR. PETTI: Yes.

19 MEMBER CORRADINI: So in the original
20 TRISO particle you've got the kernel, the silicon, no
21 I guess you have a kernel, a buffer layer and then
22 the silicon carbide and another buffer layer.

23 MR. PETTI: No, no, no. A kernel, a
24 buffer, an inner pyro carbon layer, a silicon carbide
25 and an outer pyro carbon layer.

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1 MEMBER CORRADINI: Thank you. So which
2 is duplicative that could be lost? In other words if
3 tomorrow you were to say I want to simplify the
4 process. You've made it clear that silicon carbide
5 is important. So what would you take off if you put
6 this in a binder or compact? The outer pyrolyzed
7 carbon layer? This is an off the wall sort of
8 question.

9 MR. PETTI: They're all there for a
10 different, they're each there for a reason.

11 MEMBER CORRADINI: So they're all
12 critical, nothing is removable?

13 MR. ALBERSTEIN: Necessary and
14 sufficient.

15 MR. PETTI: Yes, the inner layer is there
16 because the chemicals that are used to make silicon
17 carbide, chlorine can attack the kernel so the pyro
18 carbon's there to protect the kernel. The outer pyro
19 carbon is there so that the matrix has something to
20 grab to. And I think it would be harder to grab to
21 pure silicon carbide. That's what I would get rid
22 of.

23 MR. ALBERSTEIN: It also keeps the
24 silicon carbide in compression under irradiation.

25 MR. PETTI: But I do not believe that was

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1 designed that way. I think that was a, look at that.
2 The big benefit is that the pyro carbon's shrink and
3 keep that silicon carbide in compression.

4 MEMBER CORRADINI: So what you're telling
5 me, I'm a little bit off base. But let me ask and
6 then you'll tell me go away, we'll talk later. So
7 what you're telling me is you have a recipe. The
8 recipe works. Don't screw with the recipe because it
9 works.

10 And the understanding of taking something
11 off has an effect, I'm most interested in the
12 pyrolyzed carbon layers, not the buffer, not the,
13 because you've made it very clear what the SiC does.
14 So if I took out something could you predict the
15 effect or you'd have to test the effect?

16 MR. PETTI: It depends on what it is. If
17 it's in the particle, I can predict the effect. I
18 can tell you what happens when you lose the outer
19 pyrolyzed carbon.

20 MEMBER CORRADINI: You could?

21 MR. PETTI: Yes, but if you want to put
22 a different matrix or something that's different.

23 MEMBER CORRADINI: No, I have a question,
24 something came to my head.

25 MR. PETTI: Right. But the nature of how

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1 the outer pyro carbon adheres to the matrix, you
2 know, they're both carbon so it works. So, you know.

3 MEMBER REMPE: Wasn't there an example
4 with the NPR where they added another layer and that
5 didn't work so well too?

6 MR. PETTI: Yes, the program is very,
7 sort of cautious about just trying stuff.

8 MEMBER CORRADINI: So let ask the final
9 off-the-wall question. So if the pyrolyzed carbon
10 layers were reduced in size, is this the minimum
11 thickness or is this just from a recipe standpoint
12 the acceptable thickness?

13 MR. PETTI: You might be able,
14 particularly on the outer you might be able to go
15 thinner.

16 MEMBER CORRADINI: Fine, okay, I'll stop.

17 MR. PETTI: I think so. The inner --

18 MEMBER CORRADINI: You and I had talked
19 about other things before so I'm --

20 MR. PETTI: Right but he's thinking, he's
21 in a completely different sphere.

22 MEMBER CORRADINI: It was just a
23 question.

24 MR. PETTI: The other thing is that if
25 you look at the cesium release that we can measure in

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1 the matrix, it's very consistent with the diffusion
2 coefficients that we got from the Germans. So that
3 is conservative. So we'll continue to look at this.
4 But right now we see no evidence of any degradation.
5 Although a priori one might think with all of this,
6 this is one of the great surprises of AGR-1 that
7 we've moved a lot of this material outside of silicon
8 carbide and it just looks absolutely fine.

9 So okay. So --

10 MEMBER CORRADINI: Do you know why?

11 MR. PETTI: We're getting there. But
12 we're in the middle of PIE. So I think we're really
13 trying to get to a new level of understanding of the
14 performance and the transport. In terms of the mass
15 balance we're looking at silver, cesium, strontium,
16 europium, cerium, palladium. We do look for
17 ruthenium, Tom, but we just don't see it.

18 We're characterizing the microstructures
19 at both the micro and the nanoscale. No palladium
20 corrosion or attack has been observed. But the
21 models would predict we should have seen significant
22 amount. And they conservatively, the models I can't
23 show you all this today given the time,
24 conservatively overpredict how much cesium you get
25 under normal operation.

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1 DR. KRESS: Does that worry you?

2 MR. PETTI: No, in the end I think this
3 is because if you look at the old German silicon
4 carbide, you can see little bits of porosity.

5 DR. KRESS: You know, when I see
6 something like that I worry that my model's not
7 correct. You know, it's a concern.

8 MR. PETTI: Yes, no, I think it just has
9 to do with the microstructure is just so much better
10 material. That's what I think.

11 So we've done accident heatup testing.
12 This is the real test of the fuel where you saw the
13 plot on the right. We basically put the fuel in the
14 furnace for hundreds of hours at 16, 17 and 18
15 hundred. We've completed five tests at 16 and 17.
16 And we're going to do 1800 this year.

17 We're actually going to do one that
18 mimics that blue curve, a time temperature curve
19 instead of a constant because there's concerns about
20 that. That's on the plate for this year. And we're
21 seeing very, very low release. What we're seeing
22 released is in fact material that diffused into the
23 matrix under irradiation. There is no release from
24 intact particles. This is absolutely stunning
25 result.

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1 The silicon carbide is this good. So to
2 test that because you're really looking at small
3 numbers is we will deconsolidate and test a bunch of
4 particles and just show that's the case. So this is
5 very, very interesting in terms of the safety
6 testing. And here's my one plot where I compare now
7 the releases in these high temperature heatups
8 against what a diffusion model would predict taking
9 our best estimate diffusion coefficients.

10 So all the ones with data points are the
11 experiments. And the blue line is the prediction of
12 just what you'd expect from pure diffusion through
13 the particles. And you can see that we overpredict,
14 particularly in the strontium. The purple line is a
15 defect, there's one defective particle. If you look
16 at the release of cesium, the purple line and
17 strontium the purple line, they're the same. They're
18 $2.4 \cdot 10^{-4}$. They're roughly one particles inventory.

19 So we then said well we can do that. So
20 we shut off all the diffusion in the model and we've
21 calculated one defective particle and that's the
22 solid green. And you can see that we underpredict.
23 And so that's not a huge surprise. These are all UO2
24 correlations so it looks like maybe UCO releases
25 cesium a little bit greater. And the strontium the

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1 chemistry is very different in UCO. It doesn't come
2 out of UO2 but we do expect that there could be a
3 little mobile in UCO.

4 So this is a snap shot. But will the
5 real, the reactor vendor codes sort of predict are
6 these blue lines.

7 MEMBER CORRADINI: Remind me. PARFUME is
8 going to be what you would recommend to the owner,
9 operator as their fuel?

10 MR. PETTI: Not necessarily. I really
11 don't know what --

12 MEMBER CORRADINI: The PARFUME is the
13 currently accepted way to do an estimate. And since
14 I don't remember, you told us this two years ago in
15 a meeting and I can't remember.

16 MR. PETTI: This is, yes, I mean the
17 vendor's going to do what the vendor's going to do.
18 This is perhaps the most complex, the most detailed
19 code. But it may be impractical to do something like
20 this at reactor scale because it's so detailed.

21 MEMBER CORRADINI: Okay.

22 MR. PETTI: But again, so there's lots of
23 margin it looks like because of course we're not
24 releasing anything. So in terms of our
25 accomplishments we're nearing completion on the

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1 safety testing for AGR-1. We'll finish that up this
2 fiscal year demonstrating real robustness. Again,
3 very low releases after hundreds of hours, no
4 particle failures and no noble gas release measured,
5 which means you have not failed all the layers.

6 So we do see, when we see cesium it's
7 usually indicative of a defect. We go look, we find
8 it. It's a defect. Again, the releases are
9 associated with fission products that have diffused
10 into the matrix. No diffusive released from intact
11 particles is what we believe is going on right now.

12 So in terms of that failure fraction
13 specification it's six times 10^{-4} . We've not tested
14 enough fuel to make a statistical statement. But I
15 believe if we test enough, we may end up with a zero
16 out of how ever many we test and what that means
17 statistically. So we think that we'll be able to do
18 that.

19 We still need the data on the water and
20 air ingress. That's in the plan. And the historical
21 database on the diffusion coefficients seemed to
22 overpredict the measured releases. And I think
23 that's largely because the silicon carbide is just
24 better than what the Germans made.

25 CHAIR BLEY: At one point, I think it was

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1 talking about kernel migration, Tom had asked you
2 what, how much carbide you need. And you said it
3 wasn't real precise. There are other characteristics
4 that are effected by the carbide, some of which you
5 just showed us. Is there range or did you just pick
6 one and it's worked really well or have you got,
7 there's a minimum you need or is there a maximum that
8 gets you in trouble with cesium?

9 MR. PETTI: Yes, so there's a, I mean
10 what you do is you look at the thermochemistry and if
11 you put in too much carbon then too many of the
12 lanthanides get mobile. If you don't put enough you
13 don't tie it up, the chemistry doesn't work. And so
14 I think our spec is like 20 to 40 percent and we just
15 hit --

16 CHAIR BLEY: Okay. So it's fairly broad,
17 but it's --

18 MR. PETTI: Fairly broad. But and it
19 depends on the burnup you want to go to. So in the
20 old days in the NPR where we had HEU fuel you changed
21 the spec a little bit to expect the higher burnup and
22 the greater number of fissions. So, yes, this was
23 all developed, you know, 25 years ago. It's just
24 taken us this long to prove what was very compelling
25 on paper for thermodynamics that it works.

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1 So we'll complete our safety testing in
2 PIE this year including the safety testing at 1800.
3 And as Don knows at the last gas reactor conference,
4 1800 is when the Germans began to see degradation.
5 I'm not convinced. We did 1700 and saw no
6 degradation. We may not see degradation at 1800,
7 which again would be very encouraging.

8 We'll complete AGR-2 this fiscal year,
9 complete AGR-3/4 in 2014. And then do the follow on
10 safety testing in PIE in 2014 and 2015. AGR-5/6/7,
11 the qualification and margin testing is scheduled for
12 2016. The follow on PIE campaign 2018 to 2020. And
13 then AGR-8 follows beyond that.

14 So in one slide sort of what's the key
15 results in terms of fabrication. I think we
16 understand the process much better. We brought
17 today's technology to it. We've improved fabrication
18 and characterization by the vendor. We've had
19 outstanding irradiation performance of a large,
20 statistically significant population under high
21 burnup, high temperature HTGR conditions. We've
22 confirmed the expected superior radiation performance
23 of UCO at the high burnup.

24 The PIE indicates a lot of silver release
25 because we ran the experiment very hot. But it's

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1 consistent with the models. They can calculate
2 silver reasonably well. No cesium released from
3 intact particles under irradiation. No palladium
4 attack or corrosion despite large amounts of
5 palladium outside the silicon carbide.

6 The initial safety testing demonstrates
7 robustness of UCO TRISO under the depressurized
8 conduction cooldown condition. Low release in the
9 intact particles. All the releases to date
10 attributed to defects or fission products that were
11 released under irradiation and moved in the furnace.

12 MEMBER CORRADINI: The last bullet, why
13 are you saying under or depressurize, excuse me? I
14 misread that.

15 MR. PETTI: And then no failures to date.
16 So we are learning some very new things in terms of
17 what we think the real limits of UCO TRISO are.
18 Today we don't really know. So there's a lot we're
19 going to learn here. So in summary, we're providing
20 the data necessary to understand the behavior for the
21 modular HTGR.

22 We're laying the technical foundation
23 needed qualify the fuel made to process product
24 specifications within the envelope of the operating
25 and accident conditions that bound modular HTGRs.

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1 Our results to date are consistent with the current
2 design assumptions that are being made. And we're
3 obtaining additional data to support the development
4 and validation of models.

5 And our results to date are generally
6 consistent with the safety design basis, including
7 the functional containment and mechanistic source
8 term approaches presented today. So it is all sort
9 of lining up and fitting together.

10 CHAIR BLEY: Very good. Any more
11 questions from the Committee? And that was the last
12 item on our agenda.

13 MS. BANERJEE: Public, are we having --

14 CHAIR BLEY: Well I was going to do that
15 now. But if we can open the phone line. But I'll
16 ask inside now.

17 FEMALE PARTICIPANT: I can go and check.

18 CHAIR BLEY: Okay. Are there any
19 comments from any members of the public here in the
20 room?

21 MR. SHAHROKHI: I'd like to make a short
22 statement.

23 CHAIR BLEY: Please come to the mike and
24 identify yourself.

25 MR. SHAHROKHI: My name is Farshid

1 Shahrokhi. I work for AREVA US in Lynchburg,
2 Virginia. Today I'm representing the NGNP industry
3 alliance. The alliance is an organization, it's a
4 501(c), a nonprofit organization. Our current
5 members are about 13 or 14 members. It includes end
6 users, operators and designers and suppliers.

7 And we've been involved with the NGNP
8 project. In fact we've just signed a public, part
9 private partnership with DOE. As our first task is
10 to perform an economic analysis and some trait
11 studies on our selected design which is the AREVAs
12 steam cycle high-temperature gas-cooled reactor.
13 It's a 625 megawatt prismatic design, two-loop, very
14 similar to some of the pictures that you've seen.

15 It's a larger, higher power reactor, but
16 it is a prismatic design. You use the compacts fuel.
17 We've been involved with the NGNP project since its
18 inception. Some of our members have been supporting
19 the NGNP project. And we are hoping that the, we're
20 closely following the interaction, these, this
21 generic licensing interaction.

22 And we have put together a business plan
23 and we're trying to capitalize the development
24 venture of the business plan which says we are
25 talking with investors to get us going to begin the

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1 design of this business plant. No one company or can
2 afford to design this reactor or bring it into a
3 commercial role so we are looking for investors. And
4 the investors and member, organization members are
5 looking for clarity, licensing clarity and
6 continuation of the fuel qualification which is
7 really at the crux of this technology. Thank you.

8 CHAIR BLEY: Thank you. And --

9 MS. BANERJEE: Theron is opening the line
10 up for public comment. So you can ask again.

11 CHAIR BLEY: Any other comments in the
12 room? We think we've opened the phone line. Could
13 somebody out there just say you're on the line?

14 MS. BANERJEE: There were four people on
15 the line to start with.

16 CHAIR BLEY: We were expecting, well
17 there were four on the line earlier.

18 MALE PARTICIPANT: I'm on the line.

19 CHAIR BLEY: Good, thank you. Does
20 anyone on the line care to make a comment? Hearing
21 none we'll end with the comments. I'd like to go
22 around to the Members and hear anything you have to
23 say. I'll start with Mike.

24 MEMBER CORRADINI: I think, I appreciate
25 Idaho and the contractors for the DOE and their

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1 presentation. I think it was a good review since we
2 haven't heard from them I think now in a couple
3 years, at least in this venue. Some of us have heard
4 from them.

5 Yes, I guess I'd only say that I'd look
6 forward to see how the staff's, as Don was saying,
7 what the staff's roll-up of comments or their
8 assessment in how to end this off.

9 I think it's important that if there's
10 going to be some delay in activities within the staff
11 it's important that this is wrapped up in some way
12 that we get a clear idea where the staff sits on a
13 lot of these issues so that when it's picked up again
14 we don't have to revisit any of these things. And to
15 me that's very important otherwise we're going to
16 lose the certainty or at least some certainty as to
17 where to go from here relative to this, to the NGNP
18 advanced reactor. But other than that I would just
19 thank the INL and their staff.

20 CHAIR BLEY: Thanks, Mike. Charlie.

21 MEMBER BROWN: As an electrical puke I
22 was just in a learning experience trying to figure
23 out what's been going on for the last, in this
24 program. So this was a nice summary today. I did
25 appreciate the detail that was presented. I even

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1 understood some of it. Don't ask me which part
2 because I probably couldn't repeat it.

3 CHAIR BLEY: And, Tom, I know you'll give
4 us written comments. But I'd appreciate anything you
5 want to say now.

6 DR. KRESS: We'll give you written
7 comments. Maybe I ought to confess my bias. I've
8 always been an admirer of this gas-cooled reactor
9 concept. I think their approach to licensing is very
10 good. I like it very much. I like their fuel
11 concept and how they're making them.

12 I have a little bit of concern about the,
13 whether you can really show the quality. But put
14 some of those concerns to bed. I'm still glad you
15 have a monitoring system in the primary system just
16 in case. That's kind of one of my ideas of a defense
17 in depth. You know, when we were talking about
18 defense in depth.

19 CHAIR BLEY: Good example.

20 DR. KRESS: I personally don't think you
21 need any containment. So I think the staff has done
22 a real good job addressing all these questions. And
23 I like some of their questions and I like some of
24 their positions. One thing I think I had a little
25 concern with was you're obviously going to make the

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1 QHOs. I mean I can tell you that without even
2 calculating them.

3 But most of the PRAs for LWRs also
4 calculate the land contamination, the cancers and
5 effects of that nature. If you are going to release
6 some fission products and they're going to go beyond
7 the EAB and they've got things in them like the
8 linear no-threshold. And I don't know, I know that
9 there's no acceptance criteria, no regulations in
10 there.

11 But I think you ought to think about
12 those things if you're going to, somebody's going to
13 ask you about them somewhere along the line. If they
14 don't, I'm going to ask you. But I think that's the
15 one area I think you need to show that you have an
16 acceptable thing. But other than that I like
17 everything I've heard so far. I'm glad to hear the
18 update.

19 MR. ALBERSTEIN: I think one of the
20 people that might be the first to ask that question
21 would be the owner of the co-located process heat
22 using facility.

23 DR. KRESS: That might very well be. But
24 other than that I have better comments when I write
25 them down.

1 CHAIR BLEY: Okay. Thanks, Tom.

2 Charlie, you get another shot.

3 MEMBER BROWN: Yes, I just one for the
4 uninitiated for myself in listening to the defense in
5 depth aspects, that's the part I would be interested
6 in. I'm not quite as sanguine about the validity of
7 this just so super particle that will never break and
8 release nothing that you need no other more passive,
9 which is in a very active, hot environment and
10 without some type of passive, non, in other words a
11 blacksmith type technology containment of some sort.

12 And if you don't need a high pressure
13 containment under the circumstance, but a sealed
14 containment. So I'm just, I'll be interested in
15 hearing the justification in more detail on that.
16 But there's, I'm not as enthusiastic about that
17 thought process as a couple of the comments have
18 indicated today, so.

19 CHAIR BLEY: Thank you.

20 MS. BANERJEE: That's one action item
21 kind of thing we have from this meeting for --

22 MEMBER BROWN: Yes, I understand that,
23 that you noted that one. So I appreciated that one
24 being laid on the table. Excuse me, Dennis, I'm
25 sorry.

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1 CHAIR BLEY: That's all right. You have
2 a free hand here. I'd like to thank everybody as
3 well. It's been a very good day. I don't know how
4 you pack so much into the little time we allowed you.
5 But you did a bang up job of it. It was nice to hear
6 what's happened since the last briefing I had on the
7 experiments and the like.

8 And it's all very impressive and covers
9 a lot of the things I've worried about. I think that
10 this idea of coming back and talking about defense in
11 depth is probably very useful. Staff here has tried
12 several times to define defense in depth. It's out
13 there in many forms, many ways, many people have
14 tried. There are several documents have been
15 prepared over the last 20 years dealing with that.
16 They all come at it a little differently.

17 To me anything you do to lower the
18 likelihood of release or to control the amount of
19 release is beyond what you absolutely need is some
20 form of defense in depth. And certainly something to
21 cover the uncertainty aspects. But the idea that the
22 only thing that's defense in depth is a pure physical
23 barrier, well they aren't pure and they aren't
24 perfect. So you have problems with those as well.
25 So looking at the wide range of things that can do it

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1 would be very interesting.

2 In any case, thank you very much again.
3 I appreciate you all coming and your answers to all
4 the questions. And at this point we'll adjourn the
5 meeting.

6 (Whereupon, the hearing in the above-
7 entitled matter was concluded at 2:17 p.m.)

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U.S. DEPARTMENT OF
ENERGY

Nuclear Energy

ACRS Future Plant Designs Subcommittee Meeting

NGNP Introduction

**Carl J. Sink, Program Manager
Office of Nuclear Energy
U.S. Department of Energy**

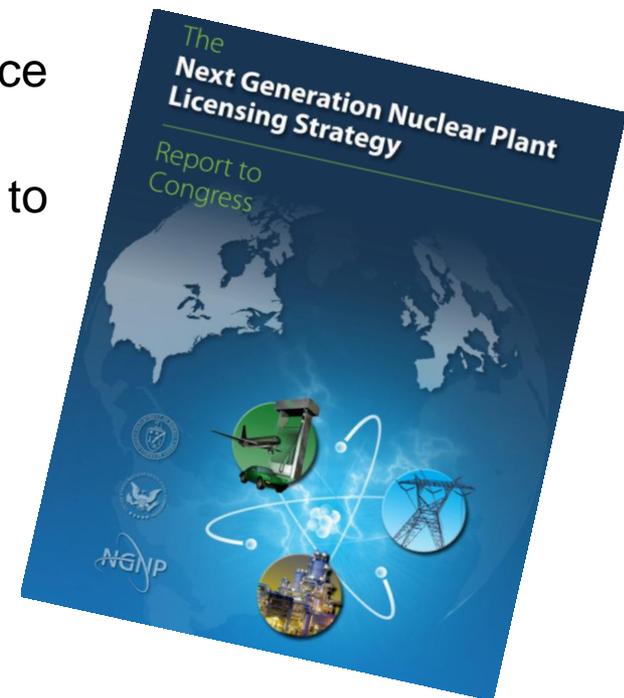
January 17, 2013



NRC- DOE Licensing Strategy – 2008 (Report to Congress)

■ **“It will be necessary to resolve the following NRC licensing technical, policy, and programmatic issues and obtain Commission decisions on these matters”**

- Acceptable basis for event-specific mechanistic source term calculation, including the siting source term
- Approach for using frequency and consequence to select licensing-basis events
- Allowable dose consequences for the licensing-basis event categories
- Requirements and criteria for functional performance of the NGNP containment as a radiological barrier



NRC- DOE Licensing Strategy – 2008 (Report to Congress)

- **The best approach to establish the licensing and safety basis for the NGNP will be to develop a risk-informed and performance-based technical approach that adapts existing NRC LWR technical licensing requirements in establishing NGNP design-specific technical licensing requirements.**
- **This approach uses deterministic engineering judgment and analysis, complemented by probabilistic risk assessment (PRA) information and insights, to establish the NGNP licensing basis and requirements.**

Continued DOE Focus on Licensing Framework

Secretary Chu letter to Congress in October, 2011 reinforces the priority that DOE places on establishing the HTGR licensing framework, based on the related NEAC recommendation

- “The NEAC also recommends that the Department continue research and development, as well as interactions with the Nuclear Regulatory Commission, to develop a licensing framework for high temperature gas-cooled reactors.”



Licensing Framework Interactions with NRC

White Paper	Submittal Date	NRC Public Meeting(s)
<i>NGNP Defense-in-Depth Approach</i> INL/EXT-09-17139	December 9, 2009	March 8, 2010
<i>NGNP Fuel Qualification White Paper</i> INL/EXT-10-18610	July 21, 2010	September 2, 2010 October 19, 2011 April 17, 2012 July 24, 2012 September 20, 2012 November 14, 2012
<i>HTGR Mechanistic Source Terms White Paper</i> INL/EXT-10-17997	July 21, 2010	September 2, 2010 October 19, 2011 April 17, 2012 July 24, 2012 September 20, 2012 November 14, 2012



Licensing Framework Interactions with NRC – cont

White Paper	Submittal Date	NRC Public Meeting(s)
<i>NGNP Licensing Basis Event Selection White Paper</i> INL/EXT-10-19521	September 16, 2010	November 2, 2010 April 16, 2012 May 16, 2012 July 10, 2012 August 22, 2012 September 19, 2012 November 14, 2012
<i>NGNP Structures, Systems, and Components Safety Classification White Paper</i> INL/EXT-10-19509	September 21, 2010	November 2, 2010 July 10, 2012 September 6, 2012
<i>Determining the Appropriate EPZ Size and Emergency Planning Attributes for an HTGR</i> INL/MIS-10-19799	October 28, 2010	January 26, 2011 November 14, 2012



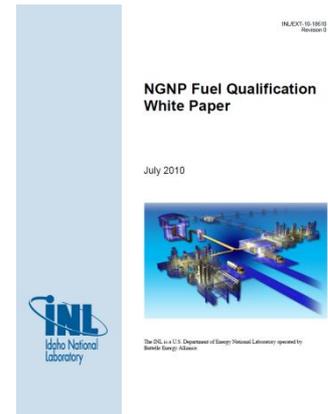
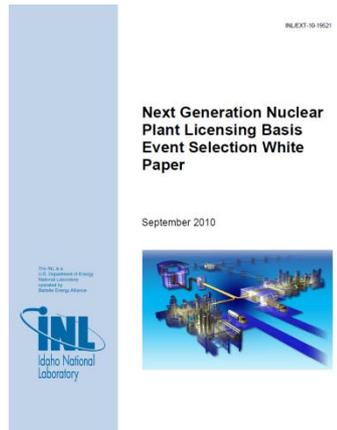
Licensing Framework Interactions with NRC – cont

White Paper	Submittal Date	NRC Public Meeting(s)
<i>NGNP Probabilistic Risk Assessment White Paper</i> INL/EXT-11-21270	September 20, 2011	April 12, 2012 September 19, 2012
<i>Modular HTGR Safety Basis and Approach</i> INL/EXT-11-22708 (submitted for information only)	September 6, 2011	None



Licensing Framework Interactions with NRC – cont

To supplement the above public meeting interactions, NGNP has also provided written responses to approximately 450 NRC Requests for Additional Information focused primarily on the topics of licensing basis event selection, mechanistic source terms, and particle fuel qualification



NRC Document

NGNP – Assessments of White Papers on:

- *Fuel Qualification and Mechanistic Source Terms*
- *Defense-In-Depth Approach, Licensing Basis Event Selection, and Safety Classification of Systems, Structures, and Components*

Transmittal Date

February 15, 2012

NRC Staff Positions Requested by DOE

- **NGNP transmitted a letter to NRC on July 6, 2012 reinforcing areas of priority for licensing framework development**
 - Consistent with focus areas summarized in NRC to DOE letter from February 15, 2012

- **NRC staff positions have been requested in four key areas**
 - Functional Containment Performance Requirements
 - Licensing Basis Event Selection
 - Establishing Mechanistic Source Terms
 - Development of Emergency Planning and Emergency Planning Zone Distances

Purpose of Today's Meeting

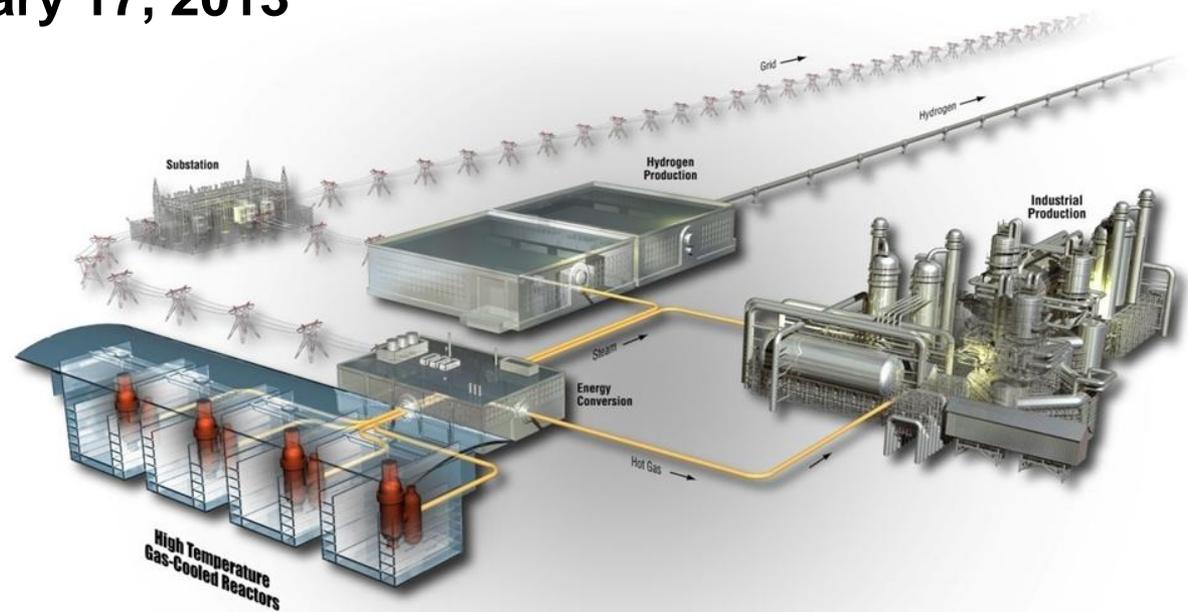
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- **Presentation topics in support of licensing framework**
 - HTGR Safety Design Bases
 - Licensing Basis Event Selection Process
 - Functional Containment Performance and Mechanistic Source Terms
 - Siting Source Terms
 - Fuel Qualification and Radionuclide Retention
 - **DOE is focused on the resolution of long-standing HTGR “licensability” issues, and establishment of key parts of the licensing framework**
 - **Eliminate the prevailing cloud of uncertainty surrounding these issues that is challenging both DOE and the private sector regarding NGNP deployment**

HTGR Safety Approach and Design Basis

ACRS Future Plant Designs Subcommittee Meeting

January 17, 2013

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Meeting Agenda

- 
- HTGR Safety Design Bases
 - Licensing Basis Event (LBE) Selection Process
 - Functional Containment Performance and Mechanistic Source Terms
 - Siting Source Terms
 - Fuel Qualification and Radionuclide Retention

Modular HTGR Safety Design Objective

- Do not disturb the normal day-to-day activities of the public



- Meet EPA Protective Action Guides at the plant boundary (EAB) for event sequences with a frequency greater than or equal to 5×10^{-7} per plant year

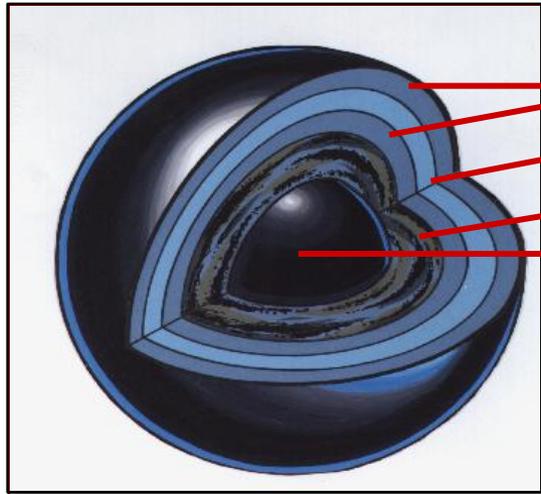
Modular HTGR Safety Design Approach

- Utilize inherent material properties
 - Helium coolant – neutronically transparent, chemically inert, low heat capacity, single phase
 - Ceramic coated fuel – high temp capability, high radionuclide retention
 - Graphite moderator – high temp stability, large heat capacity, long response times
- Develop simple modular reactor design with passive safety
 - Retain radionuclides at their source within the fuel
 - Configure and size reactor for passive core heat removal from reactor vessel with or without forced or natural circulation of pressurized or depressurized helium primary coolant
 - Large negative temperature coefficient for intrinsic reactor shutdown
 - No reliance on AC-power
 - No reliance on operator action and insensitive to incorrect operator actions

Multiple Barriers to Radionuclide Release

- Fuel Kernel
 - Fuel Particle Coatings (most important barrier)
 - Compact Matrix/Graphite
- } Fuel Element
- Helium Pressure Boundary
 - Reactor Building

HTGR Fuel



Pyrolytic Carbon (Inner and Outer)
Silicon Carbide
Porous Carbon Buffer
Fuel Kernel

TRISO coated fuel particles (left) are formed into fuel compacts (center) and inserted into graphite fuel elements (right)



Particles



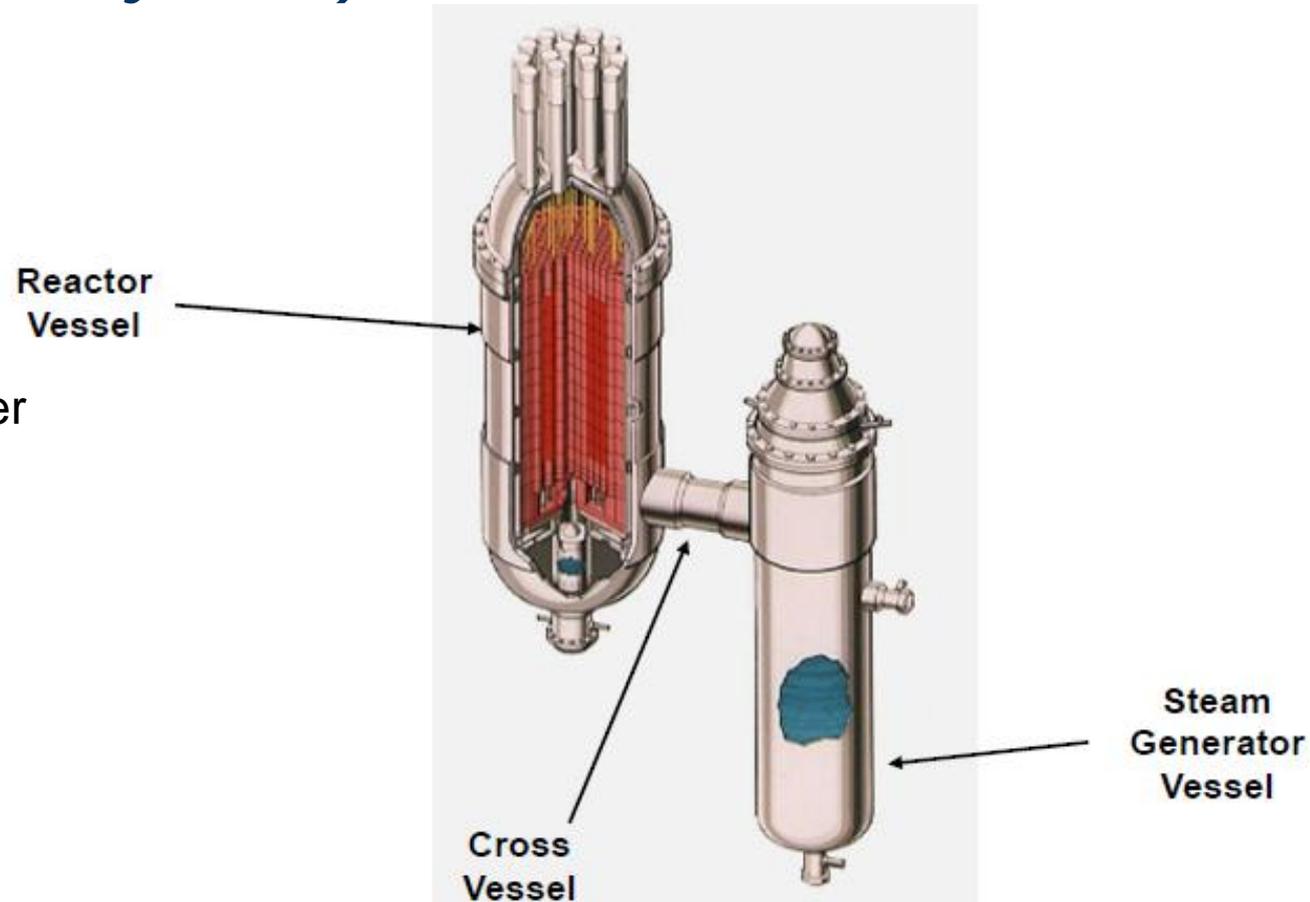
Compacts



Fuel Elements

Helium Pressure Boundary (HPB) (MHTGR Vessel System)

- ASME B&PV Code Section III pressure vessels
- Higher pressure colder helium in contact with vessels
- Loss of helium pressure does not cause loss of cooling



Reference MHTGR Embedded Reactor Building

Protects pressure vessels and RCCS from external hazards, provides additional radionuclide retention, limits air ingress following HPB depressurization

Multi-cell, reinforced concrete

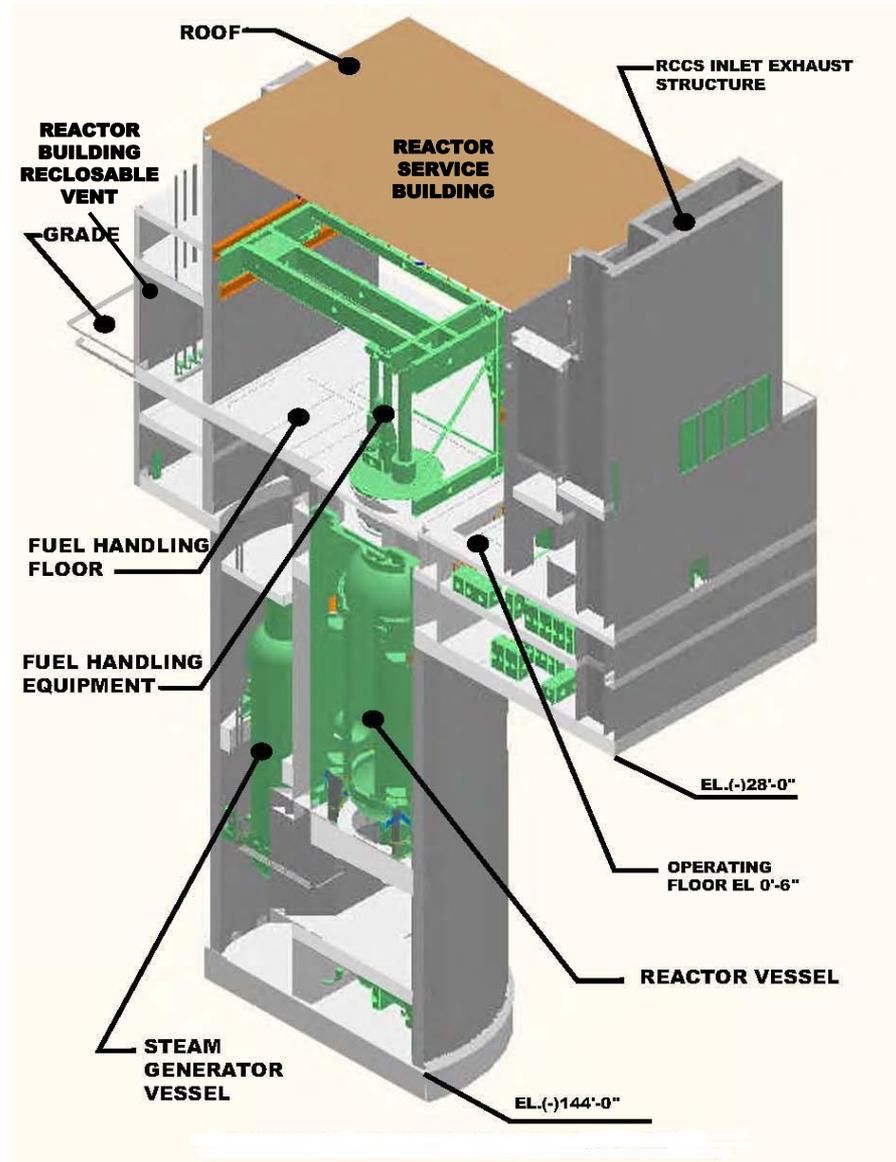
Seismic Category I

External walls ~ 3 ft thick

5 ft slab between RV and SGV cavities

Slab at grade provides
 Biological shielding
 Missile protection
 Plugs for equipment access
 Control for personnel access

Moderate Leak Rate (100% per day)



Responsive to Advanced Reactor Policy

- Use of inherent or passive means of reactor shutdown and heat removal
- Longer time constants
- Simplified safety systems which reduce required operator actions
- Minimize the potential for severe accidents and their consequences
- Safety-system independence from balance of plant
- Incorporate defense-in-depth philosophy by maintaining multiple barriers against radiation release and by reducing the potential for consequences of severe accidents
- Citation of existing technology or which can be satisfactorily established by commitment to a suitable technology development program

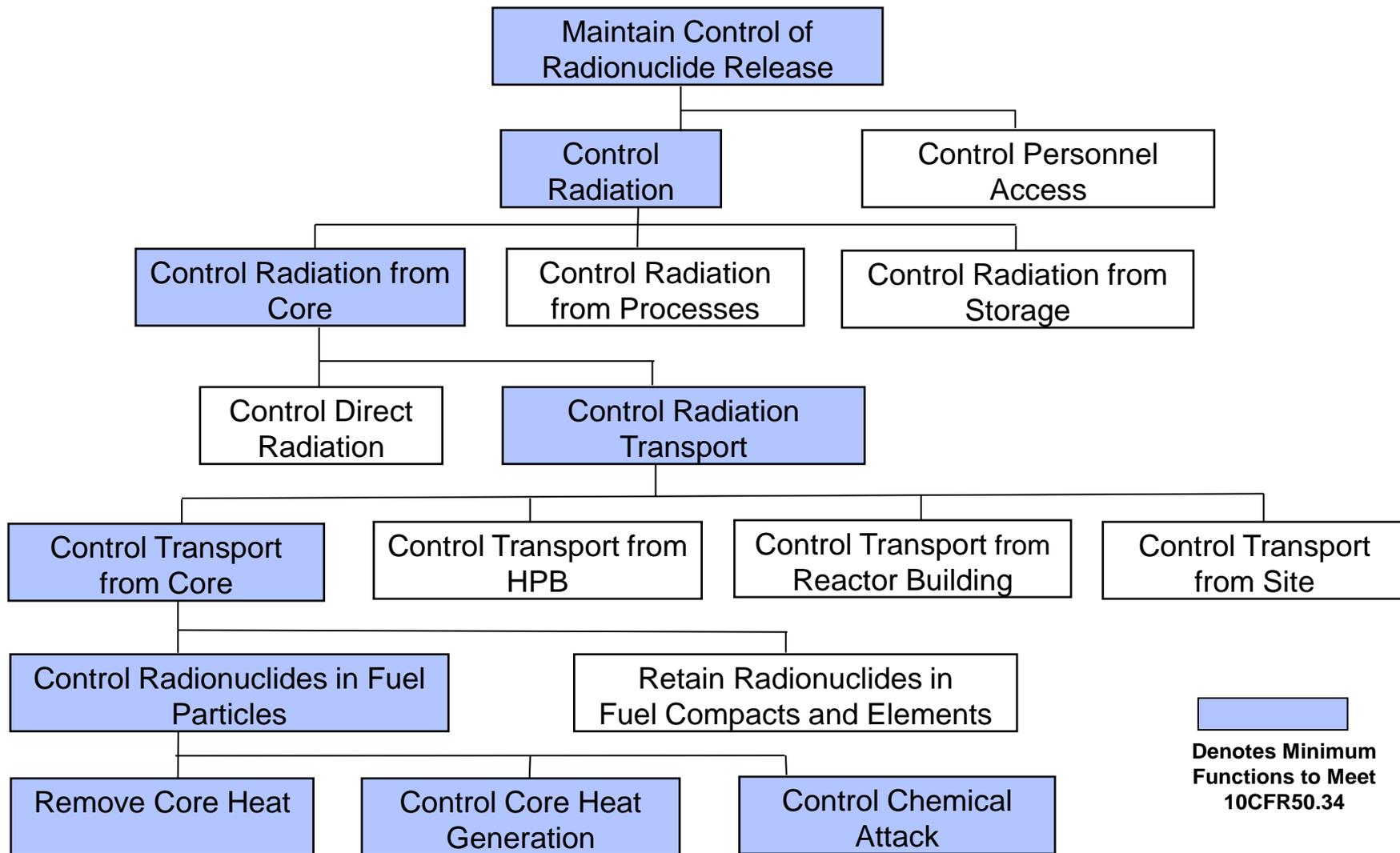
Key Element of Safety Philosophy

- Emphasis on retention of radionuclides at source within fuel means:
 - Manufacturing process must lead to high quality fuel
 - Normal operation fuel performance must limit potential for immediate radionuclide release during off-normal conditions – coolant is continuously monitored during operation
 - Off-normal fuel performance must limit potential for delayed radionuclide release to a small fraction of non-intact fuel particles from manufacturing and normal operation conditions

Safety Design and Technology Development Focus

- High fuel manufacturing quality and normal operation fuel performance ensure that modular HTGR could release activity outside of fuel barriers (e.g., circulating within HPB) and stay within offsite accident dose limits
- Thus, safety design and technology development focus is on limiting incremental releases from fuel during off-normal events
- Promising AGR fuel development program results to date

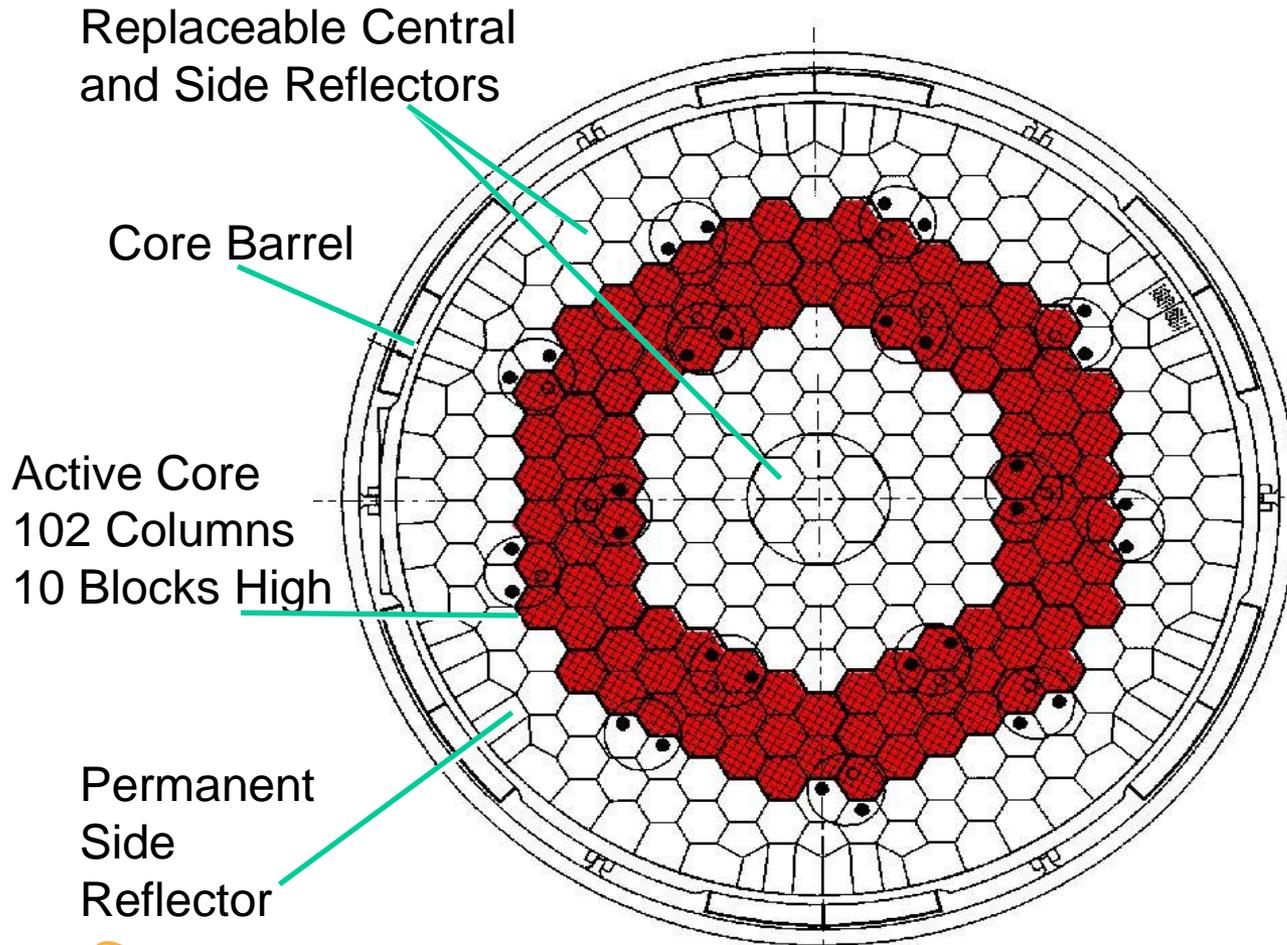
Functions for Control of Radionuclide Release



Removal of Core Heat Accomplished by Passive Safety Features

- Small thermal rating/low core power density
 - Limits amount of decay heat
 - Low linear heat rate
- Core geometry
 - Long, slender or annular cylindrical geometry
 - Heat removal by passive conduction and radiation
 - High heat capacity graphite
 - Slow heat up of massive graphite core
- Uninsulated Reactor Vessel (RV)
- Reactor Cavity Cooling System (RCCS)
 - Natural convection of air or water

Annular Core Optimizes Passive Heat Removal



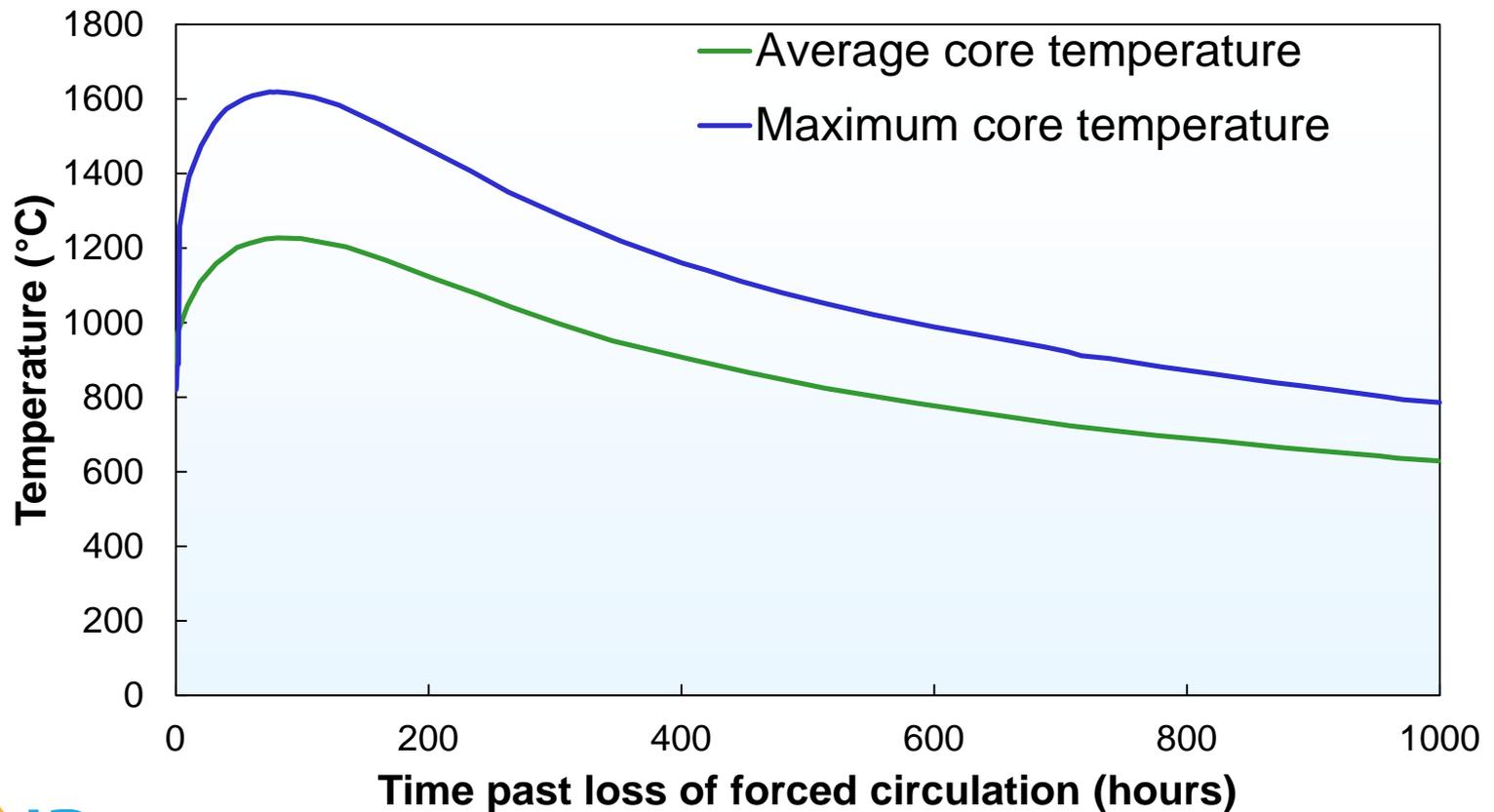
Modular HTGR utilizes annular core geometry to:

- 1) shorten conduction path
- 2) enhance surface to volume ratio

Depressurized Loss of Forced Cooling Events (DLOFCs) Demonstrate Passive Heat Removal

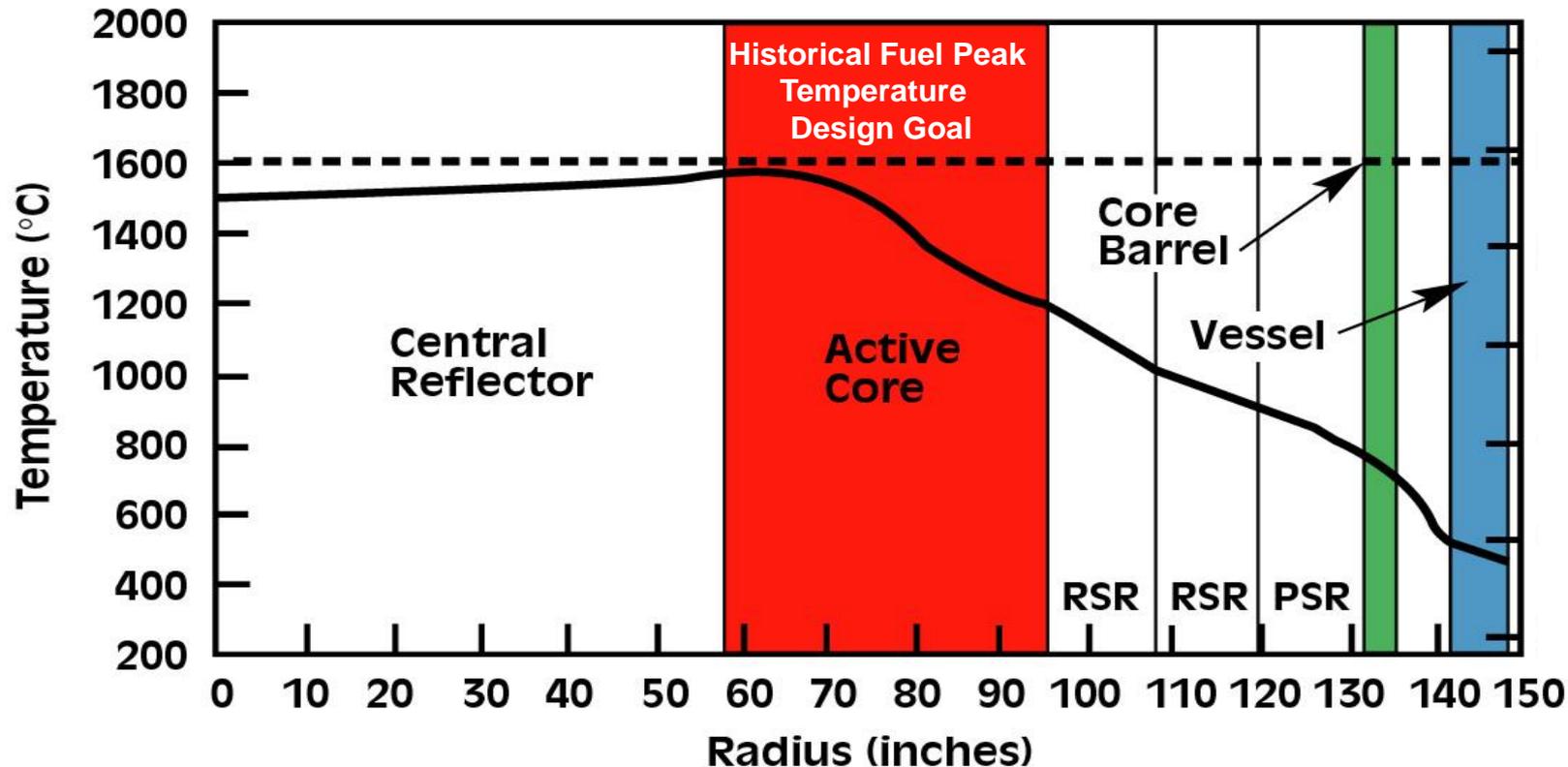
- DLOFCs are rare events in which the helium coolant is depressurized and in which the two independent forced cooling systems are both immediately and indefinitely unavailable to remove core heat
- Consequently, the core gradually heats up and the heat is removed by conduction, radiation, and convection radially to the RV to the RCCS
- DLOFCs consist of three phases that can overlap depending on the size of the leak/break in the HPB:
 - Initial depressurization (minutes to days)
 - Subsequent core heatup (~2 to 4 days)
 - Subsequent core cooldown (days)

Typical Fuel Transient Temperatures during DLOFC (MHTGR)



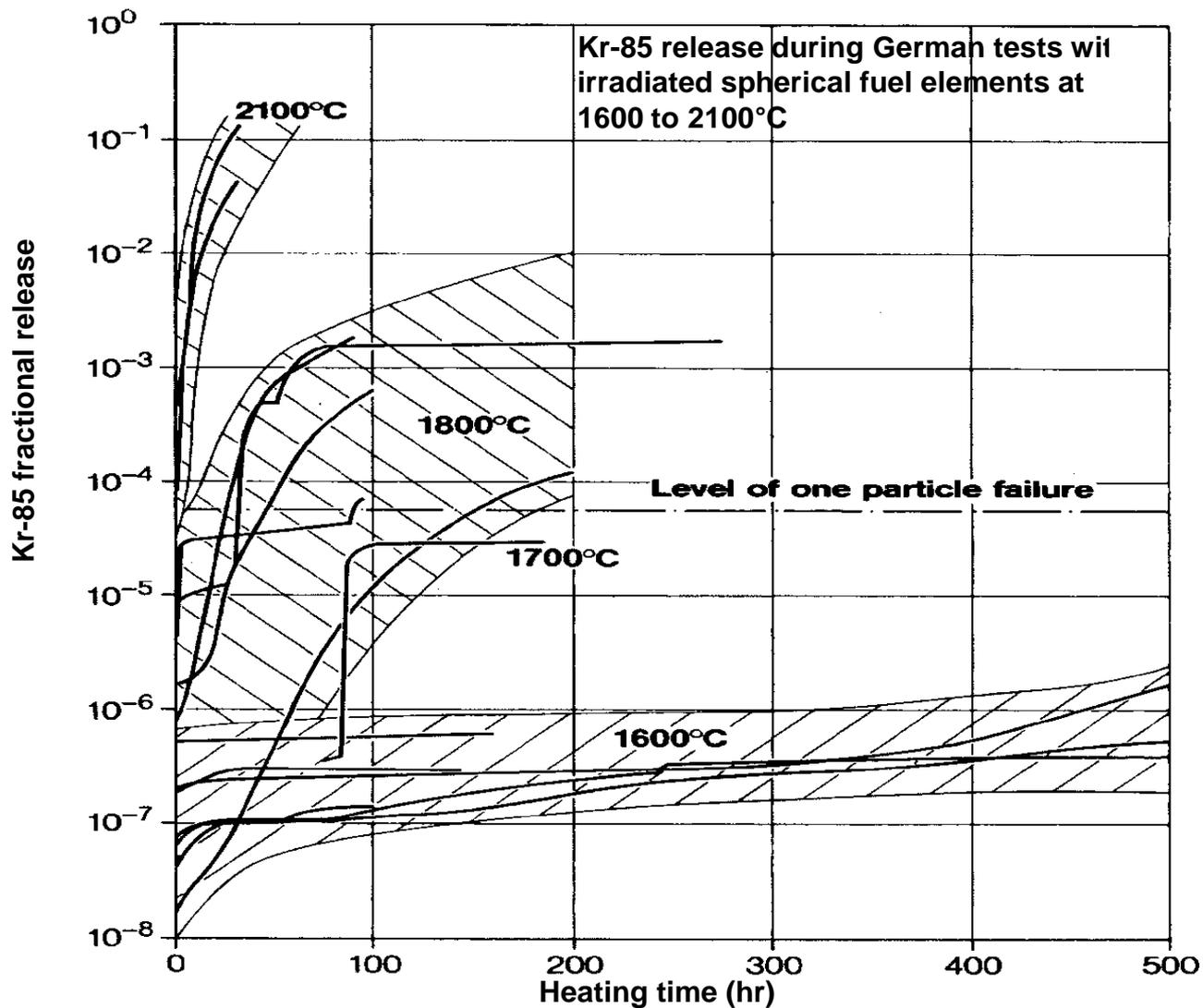
Acceptable Peak Reactor Core Temperatures at Worst Axial Location Several Days after Depressurized Loss of Forced Cooling

600 MW/102 Column



RSR: Removable Side Reflector
 PSR: Permanent Side Reflector

Fuel Particles Are Highly Retentive at 100s of Degrees Above Normal Operation



Control of Heat Generation Accomplished by Intrinsic Shutdown and Reliable Control Material Insertion

- Large negative temperature coefficient intrinsically shuts reactor down
- Two independent and diverse systems of reactivity control for reactor shutdown drop by gravity on loss of power
 - Control rods
 - Reserve shutdown system
- Each system capable of maintaining subcriticality
- One system capable of maintaining cold shutdown during refueling
- Relied on for spectrum of off-normal events, such as rod withdrawal or water ingress

Control of Air Ingress Assured by Inherent Characteristics and Passive Design Features

- Non-reacting helium coolant
- High integrity nuclear-grade pressure vessels make large break exceedingly unlikely
- Slow oxidation rate (high purity nuclear-grade graphite)
- Limited by core flow area and friction losses
- Graphite fuel element, embedded fuel compact matrix, and ceramic coatings protect fuel particles
- Reactor building embedment and vents that close after venting limit potential for gas mixture air in-leakage

Control of Moisture Ingress Assured by Inherent Characteristics and Design Features

- Non-reacting helium coolant
- Limited sources of water:
 - Moisture monitors
 - Steam generator isolation (does not require AC power)
 - Steam generator dump system
- Water-graphite reaction:
 - Endothermic
 - Slow reaction rate
- Graphite fuel element, fuel compact matrix, and ceramic coatings protect fuel particles

Safety Design Approach Summary

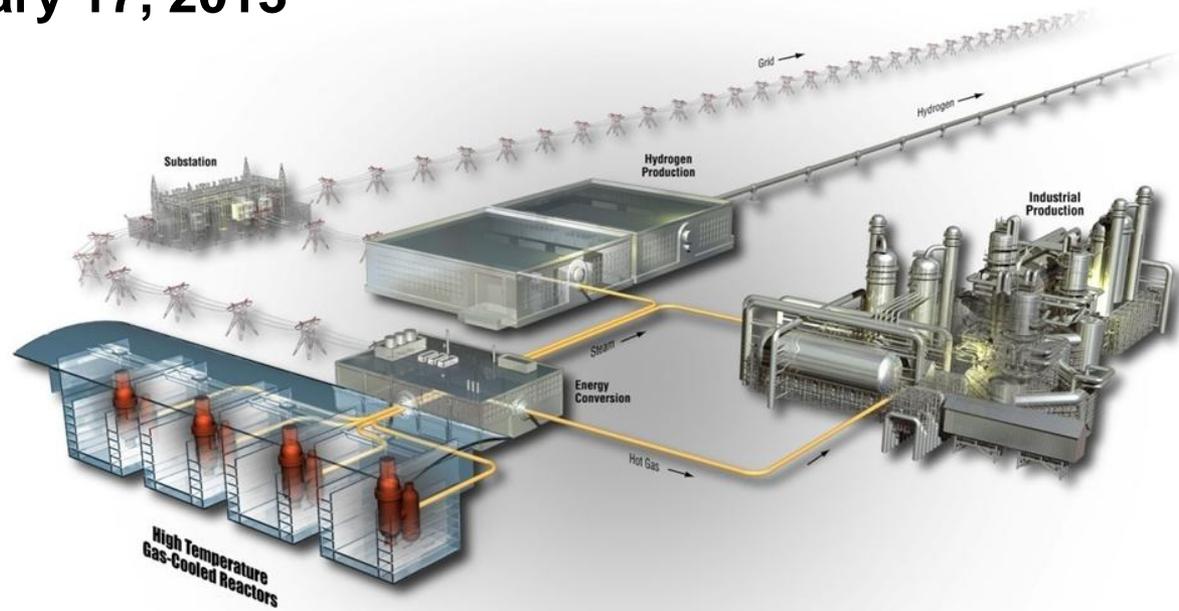
- Top objective is to meet the EPA PAGs at the EAB for spectrum of events within and beyond the design basis
- Responsive to Advanced Reactor Policy
- Modular HTGR designs employ multiple barriers to meet radionuclide retention requirements
 - Fuel Elements
 - Fuel kernels
 - Particle coatings (most important barrier)
 - Compact matrix and fuel element graphite
 - Helium coolant pressure boundary
 - Reactor building
- Retention of radionuclides at the source within ceramic fuel
 - Passive heat removal
 - Control of heat generation
 - Control of chemical attack

Licensing Basis Event Selection Process

**ACRS Future Plant Designs Subcommittee
Meeting**

January 17, 2013

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Meeting Agenda

- HTGR Safety Design Bases
- ➔ • Licensing Basis Event (LBE) Selection Process
- Functional Containment Performance and Mechanistic Source Terms
- Siting Source Terms
- Fuel Qualification and Radionuclide Retention

Outline – LBE Selection Process

- Risk-Informed, Performance-Based (RIPB) Framework and Top Level Regulatory Criteria (TLRC)
- LBE Categories and Frequency-Consequence (F-C) Curve
- Modular High-Temperature Gas-Cooled Reactor (MHTGR) Event Examples
- LBE Evaluation Structure
- Structures, Systems, and Components (SSC) Safety Classification

Requested Staff Positions – RIPB Topics

- Agree with the placement of top level regulatory criteria (TLRC) on a frequency-consequence (F-C) curve
- Establish frequency ranges based on mean event sequence frequency for the LBE event categories
- Endorse the “per plant-year” method for addressing risk at multi-reactor module plant sites
- Agree on key terminology and naming conventions for event categories
- Agree on the frequency cutoffs for the Design Basis Event (DBE) and Beyond Design Basis Event (BDBE) regions
- Endorse the overall process for performing assessments against TLRC, including issues with uncertainties and the probabilistic risk assessment (PRA), the calculational methodologies to be employed (conservative vs. best estimate), and the adequate incorporation of deterministic elements
- Endorse the proposed process and categorizations for structures, systems, and components (SSC) classification

Why Define Event Sequences Through LBE Selection Process?

- Technology neutral
- Comprehensive method for plant design and licensing to assure protection of the public for a spectrum of events
- Single failure criteria and associated redundancy may mask risk-significant accident sequences with multiple failures
- Quantitative; safety margins can be assessed

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RIPB Framework

- **What** must be met:
 - Top Level Regulatory Criteria (TLRC)
- **When** TLRC must be met:
 - Licensing Basis Events
- **How** TLRC must be met:
 - Safety Functions
 - SSC Safety Classification
- **How well** TLRC must be met:
 - Deterministic Design Basis Accidents (DBAs)
 - Defense-in-Depth
 - Regulatory Special Treatment

Bases for Top Level Regulatory Criteria (TLRC)

- Generic, technology-neutral and independent of plant site
- Quantitative
- Direct statements of acceptable consequences or risks to the public or the worker

TLRC for Protection of the Public

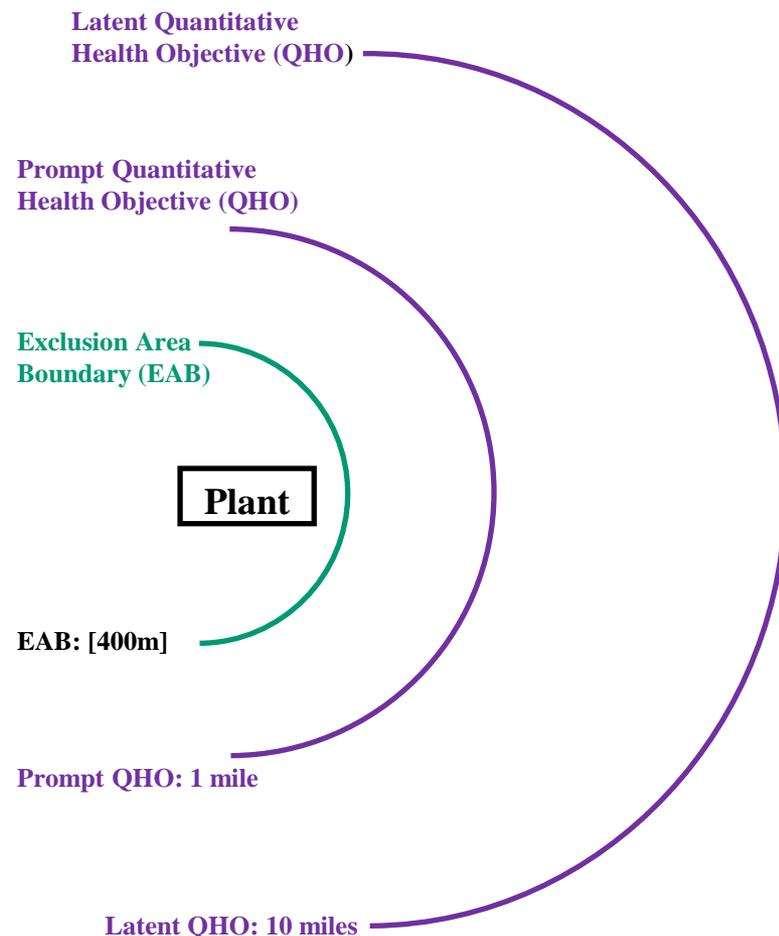
- 10CFR20 annualized offsite dose guidelines
 - 100 mrem/yr total effective dose equivalent
 - Measured on a cumulative basis annually at the EAB of the site
 - For normal operation and anticipated operational occurrences
- 10CFR50.34 (10CFR52.79) accident offsite doses
 - 25 rem total effective dose equivalent
 - Evaluated at the site EAB at 2 hour and at the site LPZ at 30 days
 - Design basis for off-normal events
- EPA Protective Action Guides (PAGs) offsite doses
 - 1 rem total effective dose equivalent for sheltering
 - Evaluated at the site EPZs
 - Emergency planning and protection during off-normal events
- NRC Safety Goals individual fatality risks
 - Prompt Quantitative Health Objective (QHO) of 5×10^{-7} /yr latent QHO of 2×10^{-6} /yr
Evaluated at 1 mile for prompt and 10 miles for latent
 - Overall assurance of negligible cumulative risks during normal operation and off-normal events

Selection of Frequency Axis for TLRC Placement

- Use of a risk assessment process leads to a frequency-consequence (F-C) curve construct
- Event likelihood is implicit in the current regulations; however, event frequencies are not typically stated
- **Event sequence frequency** is used since it is the frequency to be compared to the doses of the TLRC and the frequency for the NRC safety goal QHOs that are expressed as risks
- **Mean frequency** is selected as the best single measure of the expected outcome
- Event frequencies are expressed on a **per plant year basis**:
 - This is the important measure to the public (not whether a radionuclide release originated from one particular reactor module or system)
 - Provides the flexibility for the consequence limits in the TLRC to be met for **one or more reactor modules**

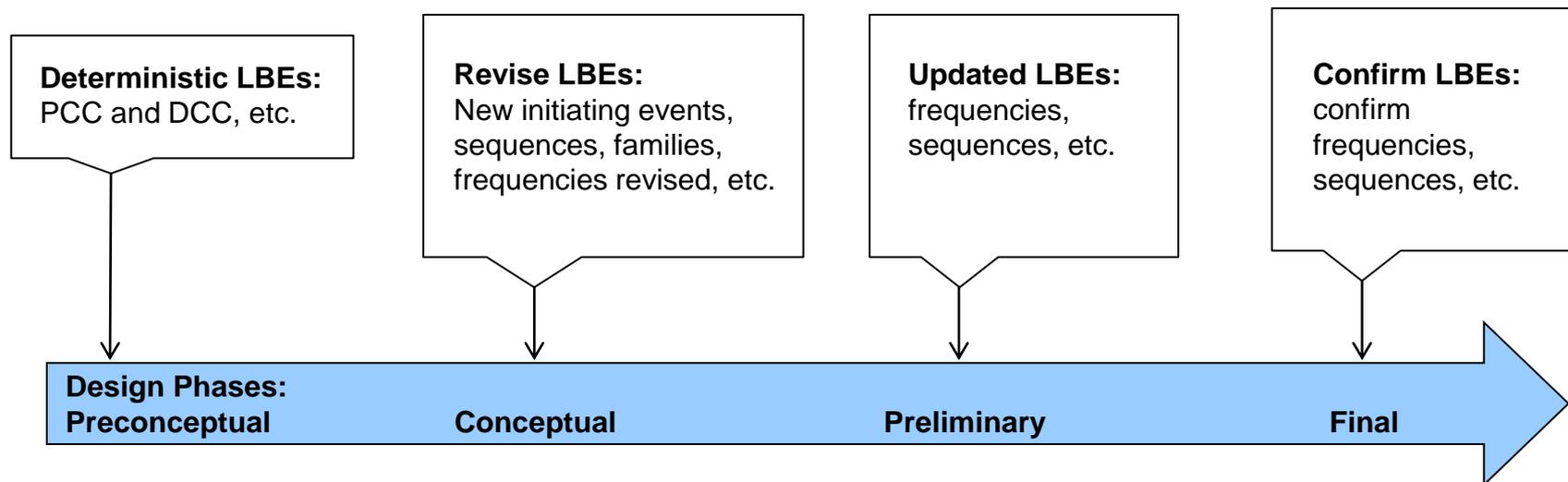
Selection of Consequence Axis for TLRC Placement

- Mean TEDE dose selected for consequence measure
- The Exclusion Area Boundary (EAB) was selected based on the following considerations:
 - It is the distance specified for the 10CFR20 and one of the 10CFR50.34 dose limits
 - Design objective is to meet the PAGs at the EAB to avoid public sheltering during off-normal events: the goal is for LPZ and EPZs to be at the same distance as the EAB (approximately 400m)
 - If met, the plant will have large margins to the average individual risk QHOs as measured within annular regions from the EAB to 1 and 10 miles, respectively
 - Supports co-location with industrial facilities



Event Selection Timeline

LBE evolution by design phase:



LBE selection process inputs vary by design phase:

- | | | | |
|---|--|---|---|
| <ul style="list-style-type: none"> • Initial design concept* • Prior HTGR experience* • Expert insights* | <ul style="list-style-type: none"> • Basic design* • Initial analyses (FMEA, scoping PRA, etc.)* • Prior HTGR experience* • Design reqmts.* • Expert reviews* | <ul style="list-style-type: none"> • Updated design* • Detailed FMEAs, etc.* • Initial PRA results* • Expert reviews* • Regulator interaction* | <ul style="list-style-type: none"> • Mature design • Detailed FMEAs, etc. • Complete PRA results • Expert reviews • Regulator feedback |
|---|--|---|---|

* **Steps actually performed during MHTGR project through early preliminary design**

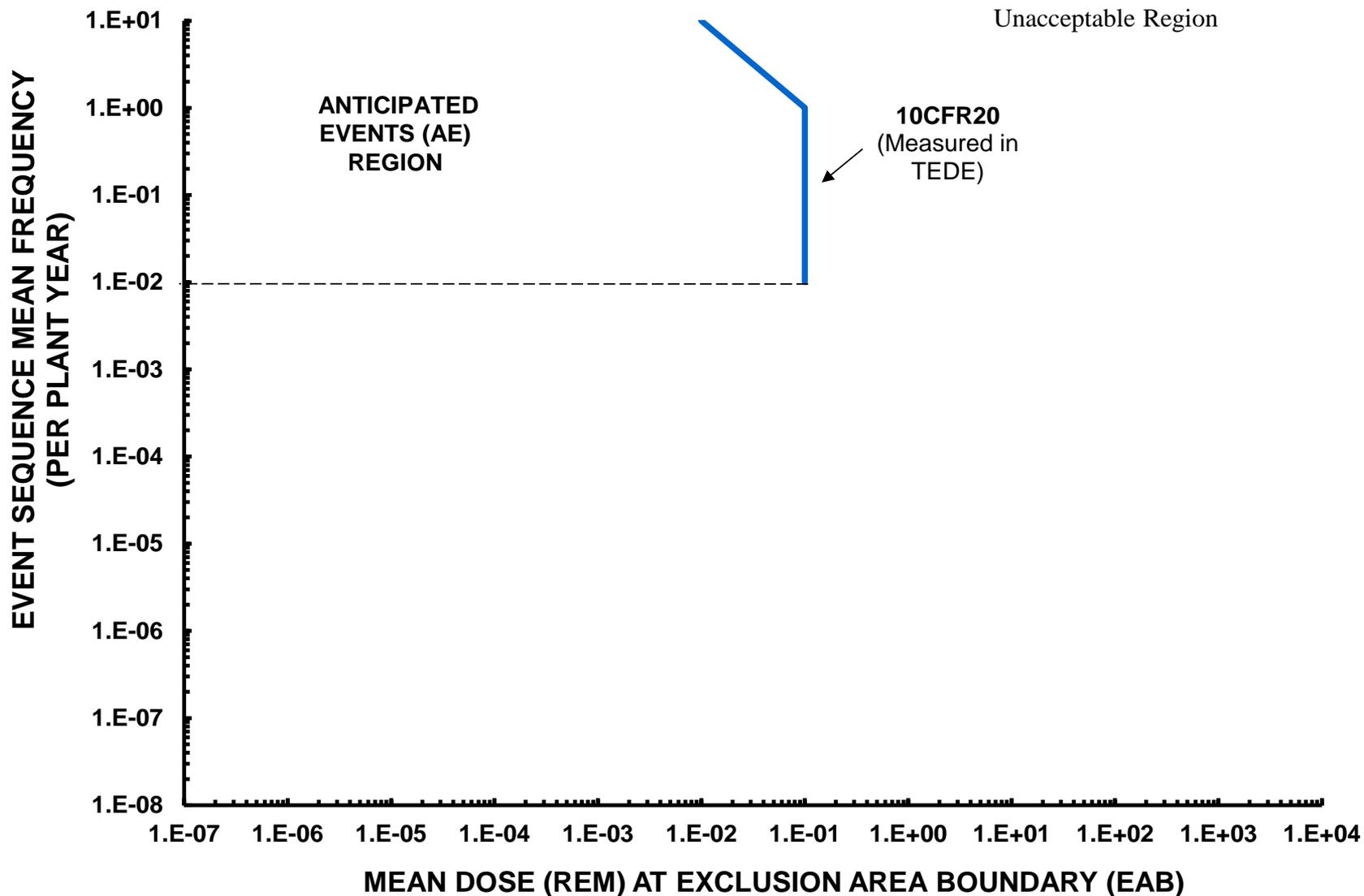
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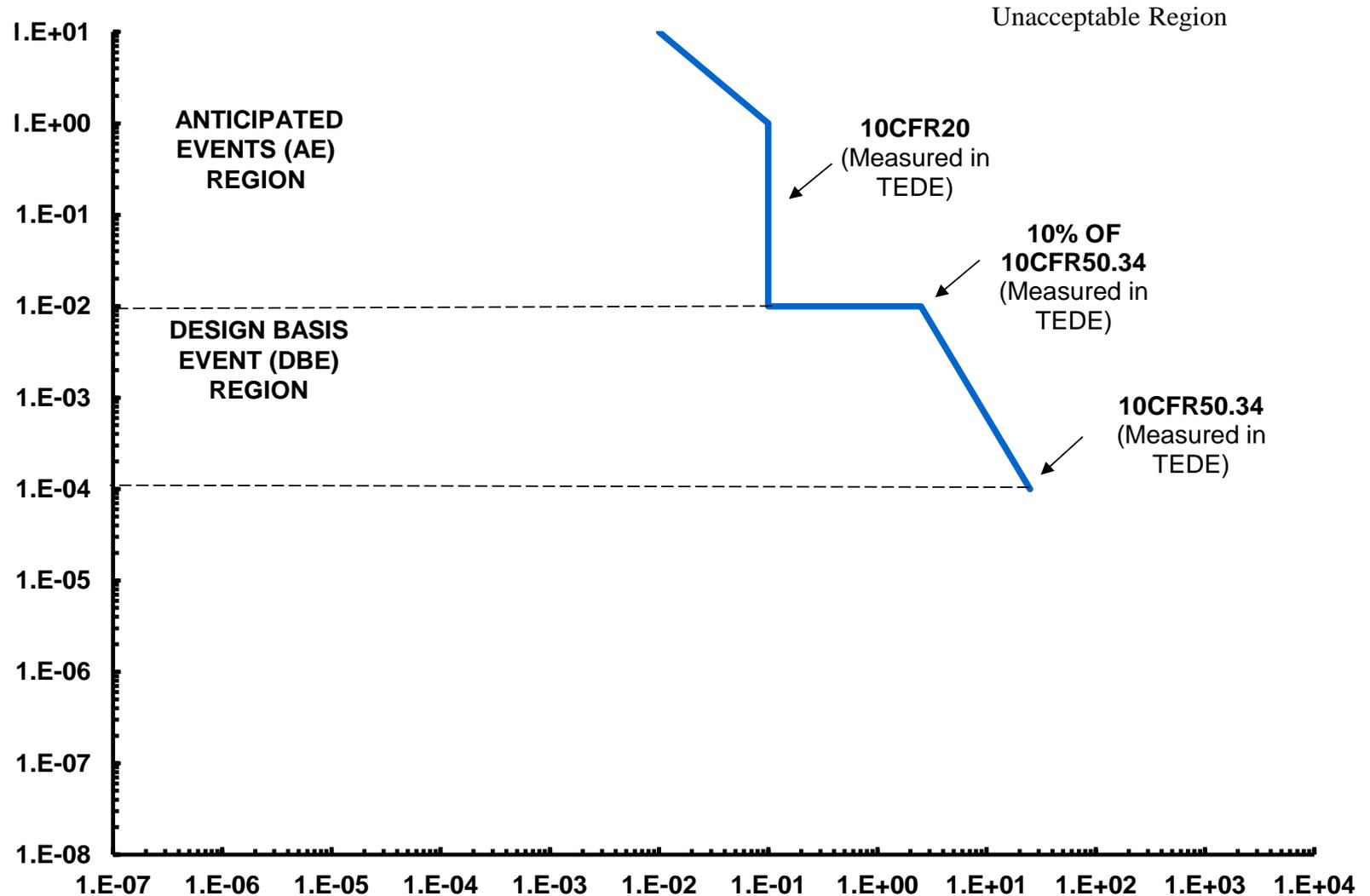
Categories of Licensing Basis Events

- TLRC apply to the full spectrum of normal operation and off-normal events
- Some specific TLRC apply to normal operation and anticipated events; others to design basis events; others to events less frequent than design basis events
- LBE categories selected:
 - Anticipated Events (AEs): $>10^{-2}$ /plant year
 - Design Basis Events (DBEs): 10^{-2} to 10^{-4} /plant year
 - Beyond Design Basis Events (BDBEs): 10^{-4} to 5×10^{-7} /plant year
 - Design Basis Accidents (DBAs)
- Design Basis Accidents (analyzed in Chapter 15 of SARs) are deterministically derived from DBEs, assuming that only SSCs classified as safety-related are available
 - The event sequence frequency for some of these DBAs are expected to fall in or below the BDBE region

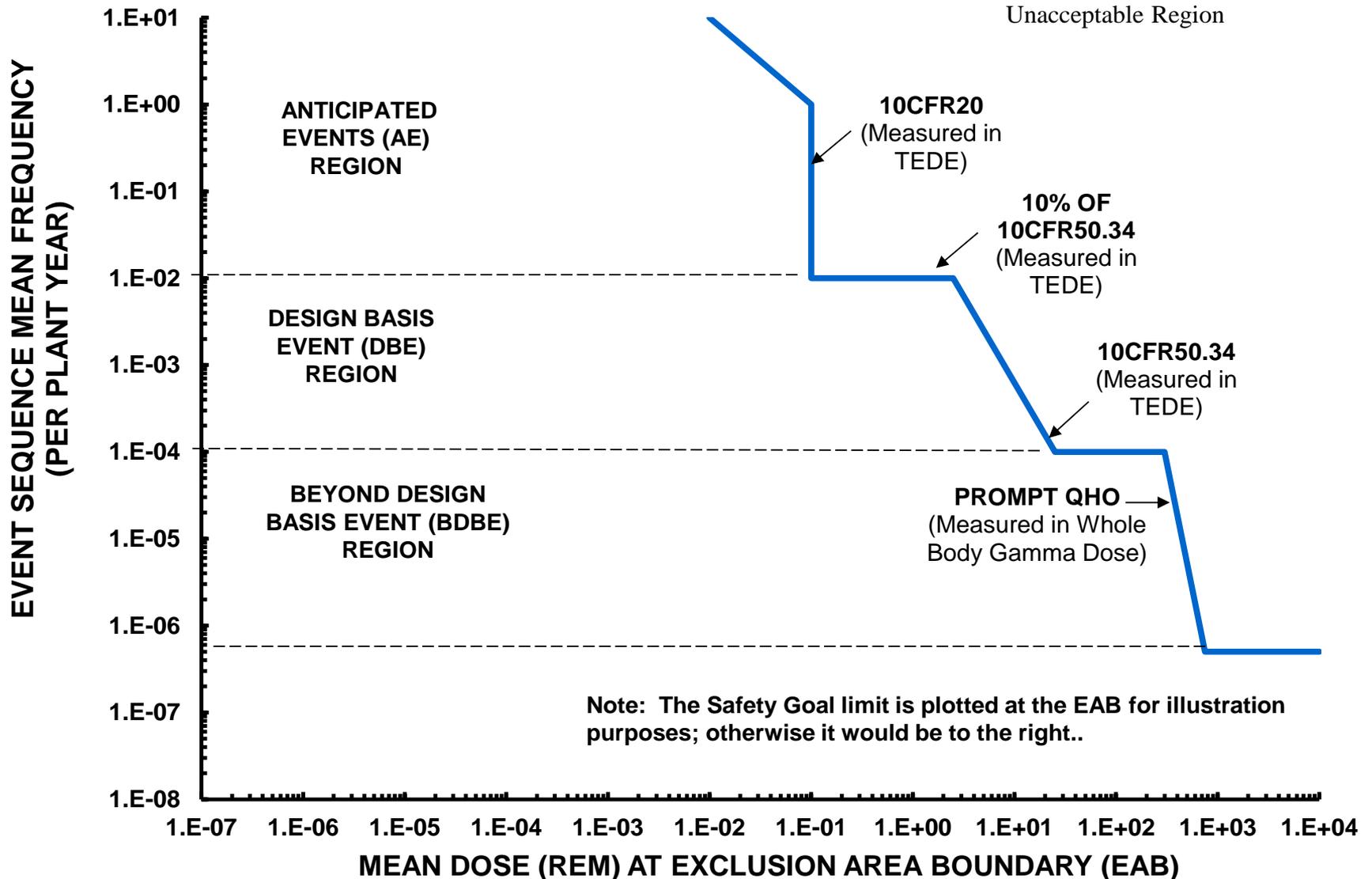
AE Region on F-C Curve



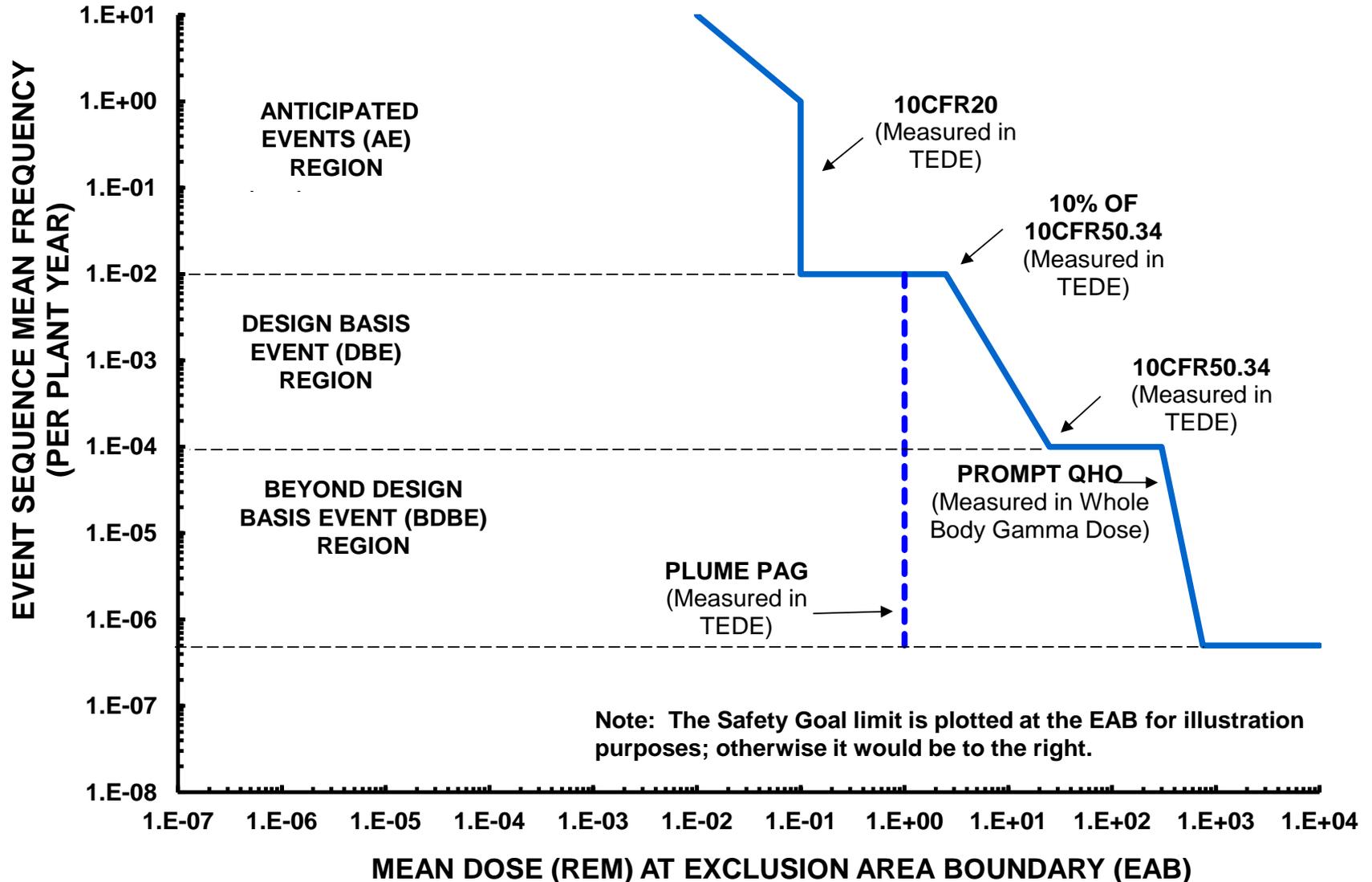
DBE Region on F-C Curve



BDBE Region on F-C Curve



NGNP F-C Curve



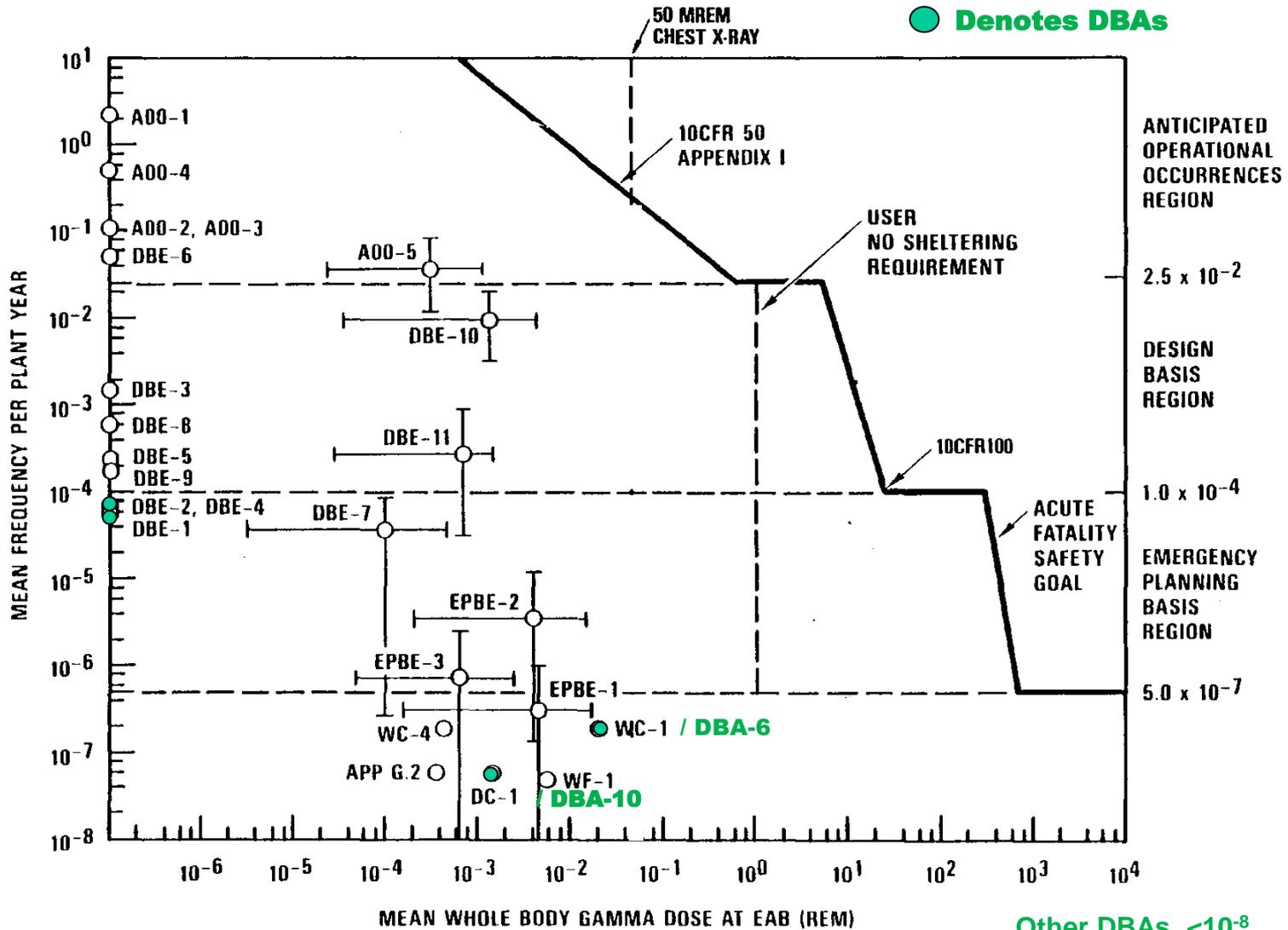
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Limiting LBEs from MHTGR

- The MHTGR PSID identified several DBEs/DBAs and BDBEs enveloped by the following highest offsite consequence DBAs:
 - DBA-6: Steam Generator (SG) offset tube rupture with SG isolation and immediate and indefinite loss of forced cooling leading to an early (min to hr) and a delayed (days) radionuclide release from Helium Pressure Boundary (HPB) via opening of Vessel System (VS) relief valve to the Reactor Building (RB)
 - DBA-10: VS relief line breach of HPB with immediate and indefinite loss of forced cooling leading to an early (sec to min) and a delayed (days) radionuclide release from HPB to RB
 - DBA-11: Instrument line leak in HPB with immediate and indefinite loss of forced cooling leading to an early (min to hr) and a delayed (days) radionuclide release from HPB to RB

MHTGR DBEs, DBAs, and BDBEs (aka EPBEs) on F-C Plot (circa 1987)



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LBE Evaluation Structure

Event Category/Type	10CFR20 – 0.1 rem	10CFR50.34 – 25 rem	EP PAGs – 1 rem	QHOs – Individual Risks
AEs	Mean Cumulative @ EAB			Mean Cumulative @ 1 and 10 miles
DBEs		Upper Bound @ EAB	Mean @ EPZ*	Mean Cumulative @ 1 and 10 miles
BDBEs			Mean @ EPZ*	Mean Cumulative @ 1 and 10 miles
DBAs		Upper Bound @ EAB		

***Design Objective: EPZ = EAB**

Treatment of Uncertainties

- The mean and upper bound consequences are explicitly compared to the consequence criteria in all applicable LBE regions
- Example of parameters considered in the treatment of uncertainty applicable to HTGR consequence analysis:
 - Fuel inventory, circulating inventory, and plateout inventory
 - Initial fraction of defective fuel particles
 - Releases from defective fuel particles
 - Reactor building deposition and leakage
- The consequence uncertainty model accounts for the release and transport of radionuclides to the atmosphere from:
 - Fuel particle kernel
 - Silicon carbide and pyrocarbon coatings of the fuel particle
 - Fuel matrix and fuel element graphite
 - Helium pressure boundary
 - Reactor building

Key Features of Modular HTGR PRA

- All sources of radioactive material addressed
- Success criteria reflect reactor's unique features:
 - Reactor specific criteria to establish safe, stable end states
 - Breaches in HPB do not result in loss of cooling
 - Need functional basis for long system mission times
 - Plant response to ATWS and SBO fundamentally different than for LWR
- Smaller number of systems to model
- Integrated event sequence model for treatment of internal and external events and all operating and shutdown modes
- Source term phenomena unique to HTGRs
- Absence of severe core damage LWR-specific phenomena
- No “core damage” or “large early release” pinch points; CDF and LERF not applicable
- Unique HTGR end states covering a range of radionuclide release categories
- Address integrated risk of multi-reactor module plant
- Address sequences to support application and ensure no cliff edge effects
- Address hazards from nearby industrial facilities

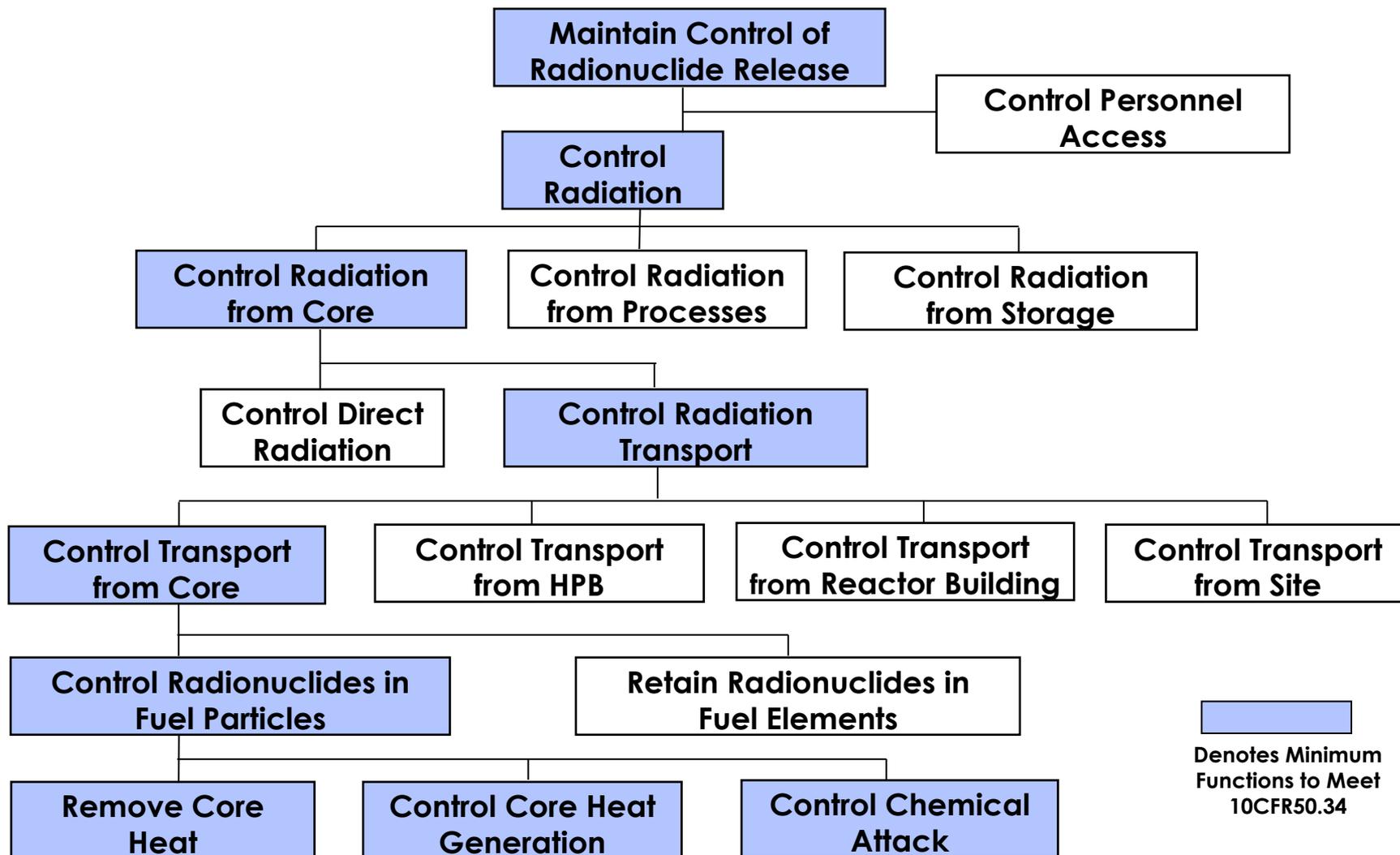
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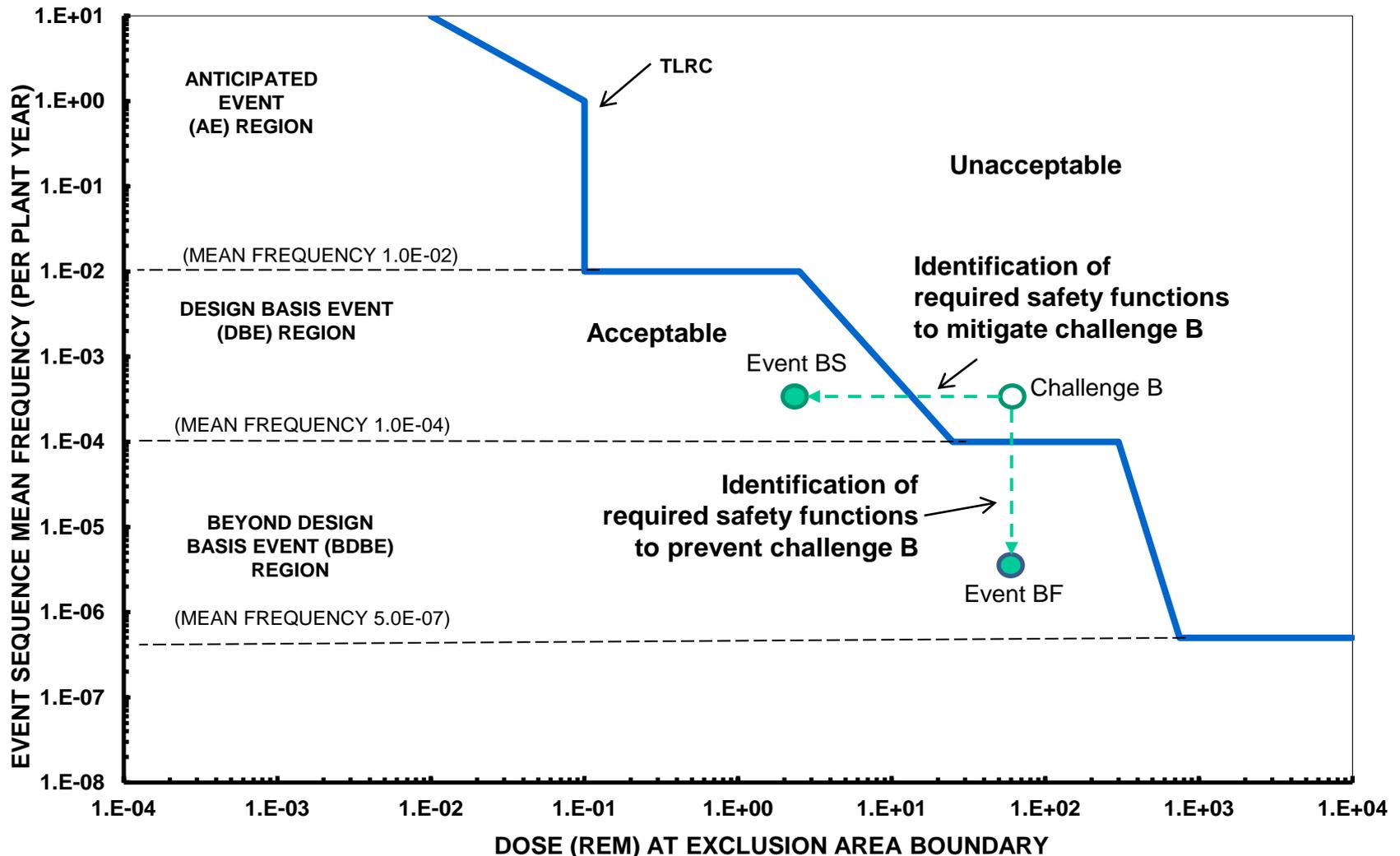
Process for SSC Classification as Safety-Related

- Determine the required safety functions for DBEs and BDBEs
- For each required safety function, determine which SSCs are available and have sufficient capability and reliability to meet the required safety function
- From this review, classify a set of SSCs as safety-related to assure that the required safety functions are accomplished

Functions for Control of Radionuclide Release



Identification of Safety Functions Leading to Safety-Related SSCs



MHTGR Example of Safety Classification for Core Heat Removal Function (1/2)

Are SSCs Available and Sufficient to Remove Core Heat in the DBE?	
Alternative Sets of SSCs	DBE 11
Initiating Event	HPB small leak
Reactor HTS ECA	No
Reactor SCS SCWS	No
Reactor RV RCCS	Yes
Reactor RV RB	Yes

MHTGR Example of Safety Classification for Core Heat Removal Function (2/2)

Are SSCs Available and Sufficient to Remove Core Heat in the DBE?

Alt. Sets of SSCs	DBE 1	DBE 2	DBE 3	DBE 4	DBE 5	DBE 6/7	DBE 8/9	DBE 10	DBE 11	SSCs Classified as SR?
IE	Transt, (LOSP +TT)	ATWS	Control rod withdwl	Control rod withdwl	SSE	SG tube rupture	SG tube leak	HPB moderate leak	HPB small leak	
Reactor HTS ECA	No	No	No	No	No	No	No	No	No	
Reactor SCS SCWS	No	Yes	Yes	No	Yes	Yes	Yes	Yes	No	
Reactor RV RCCS	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes, optimum selection to achieve capability and reliability
Reactor RV RB	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	

LBE Selection Summary

- Licensing Basis Events determine **when** Top Level Regulatory Criteria must be met
- Selected during design and licensing process with risk insights from comprehensive full scope PRA that treats uncertainties
- Include AEs (expected in life of plant), DBEs (not expected in plant lifetime), BDBEs (not expected in fleet of plant lifetimes) and DBAs (Ch 15 events derived from DBEs with only safety related SSCs available)
- Safety classification determined by examining SSCs available and sufficient to successfully perform required safety functions to mitigate spectrum of DBEs

Requested Staff Positions – RIPB Topics

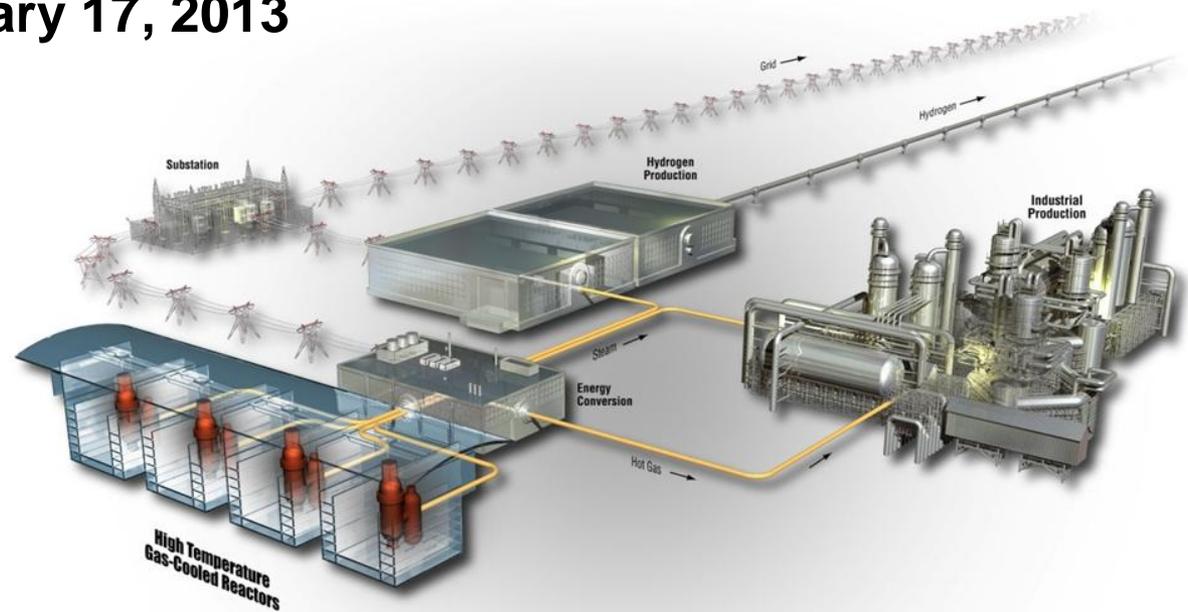
- Agree with the placement of top level regulatory criteria (TLRC) on a frequency-consequence (F-C) curve
- Establish frequency ranges based on mean event sequence frequency for the LBE event categories
- Endorse the “per plant-year” method for addressing risk at multi-reactor module plant sites
- Agree on key terminology and naming conventions for event categories
- Agree on the frequency cutoffs for the Design Basis Event (DBE) and Beyond Design Basis Event (BDBE) regions
- Endorse the overall process for performing assessments against TLRC, including issues with uncertainties and the probabilistic risk assessment (PRA), the calculational methodologies to be employed (conservative vs. best estimate), and the adequate incorporation of deterministic elements
- Endorse the proposed process and categorizations for structures, systems, and components (SSC) classification

Functional Containment Performance and Mechanistic Source Terms

ACRS Future Plant Designs Subcommittee Meeting

January 17, 2013

www.inl.gov



Meeting Agenda

- HTGR Safety Design Bases
- Licensing Basis Event (LBE) Selection Process
- ➔ • Functional Containment Performance and Mechanistic Source Terms
- Siting Source Terms
- Fuel Qualification and Radionuclide Retention

Presentation Outline

- Introduction
- Regulatory Background
- Functional Containment Performance and Mechanistic Source Term Determination
- Conclusions



- Introduction
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INL/EXT-10-17997

Mechanistic Source Terms (MST) White Paper

July 2010

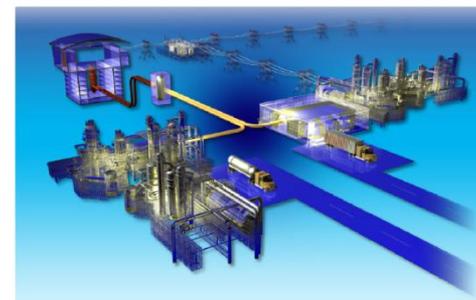
NRC ADAMS Accession Number:
ML102040260



INL/EXT-10-17997
Revision 0

Mechanistic Source Terms White Paper

July 2010



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MST White Paper Contains Information on Radionuclide Transport and Retention in the Modular HTGR

- Functional containment description
- Radionuclide behavior in the fuel, primary circuit, and reactor building
- MST models and modeling assumptions
- Sources of data on radionuclide behavior
- Experimental methods for data collection

Requested NRC Staff Positions on Functional Containment (July 6, 2012 Letter)

- Item 1.b. Establish options regarding functional containment performance standards
 - as requested by the Commission in the Staff Requirements Memorandum (SRM) to SECY-03-0047, "Policy Issues Related to Licensing Non-Light Water Reactor Designs,"
 - and discussed further in SECY-05-0006, "Second Status Paper on the Staffs Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing"

Requested NRC Staff Positions on MST (July 6, 2012 Letter)

- Item 3.a: Endorse the proposed NGNP mechanistic source terms definition – the quantities of radionuclides released from the reactor building to the environment during the spectrum of LBEs, including timing, physical and chemical forms, and thermal energy of the release
- Item 3.b: Agree that NGNP source terms are event specific and determined mechanistically using models of radionuclide generation and transport that account for fuel and reactor design characteristics, passive features, and the radionuclide release barriers
- Item 3.c: Agree that NGNP has adequately identified the key HTGR fission product transport phenomena and has established acceptable plans for evaluating and characterizing those phenomena and associated uncertainties

- Introduction
- ➔ • Regulatory Background
- Functional Containment Performance and Mechanistic Source Term Determination
- Conclusions

Regulatory Precedent for Functional Containment and Mechanistic Source Terms

- Advanced Reactor Policy Statement
- SECY Documents
 - 93-092 05-0006
 - 95-299 10-0034
 - 03-0047
 - Other SECYs discuss “new,” “revised,” or “physically based” source terms for evolutionary and advanced LWRs
- US HTGR Licensing Interactions
 - Peach Bottom (Unit 1)
 - Fort St. Vrain
 - Large HTGR (to Construction Permit stage)
 - DOE MHTGR pre-application
 - Pebble Bed Modular Reactor pre-application submittals

DOE MHTGR PSER NUREG 1338 Drafts: 1989 and 1995

- 1989 – (p 15-23) Section 15.6 “The staff has judged that the siting source term can be based on a mechanistic analysis of fuel failure and radionuclide inventory contained in the circulating helium or plated out within the primary system

Final acceptance of a mechanistically calculated source term is dependent on satisfactory accomplishment of research and development goals, satisfactory resolution of the safety issues and deferred items, and a prototype test program demonstrating that the combination of research and development findings and analytical predictions confirm the staff's detailed and overall safety conclusions for the MHTGR”

- 1995 – (p 3-16) “Commission decided that a mechanistic source specific to the design was acceptable”

DOE MHTGR PSER NUREG 1338 Draft: 1995

- 1995 – (p 4-8) “In its decision on source terms for the advanced reactors policy issues...the Commission approved the use of mechanistic source terms for the MHTGR”
- “However, the Commission criteria for use of mechanistic source terms is that the source terms had to be based on:
 - The fuel performance being well understood,
 - Fission-product transport being adequately modeled, and
 - Events considered in the development of source terms include bounding severe accidents and design-dependent uncertainties”

DOE MHTGR PSER NUREG 1338 Draft: 1995, cont'd

- 1995 – (p 4-11) “...the Commission decided that a conventional LWR, leaktight containment should not be required for advanced reactor designs. It approved the use of containment functional design criteria for evaluating the acceptability of proposed containment designs rather than the use of prescriptive design criteria”
- 1995 – (p 5-10) “[The] position regarding containment allows the acceptance of containments with leak rates that are not ‘essentially leaktight’ as described in GDC 16 for LWRs”

- Introduction
- Regulatory Background
- • Functional Containment Performance and Mechanistic Source Term Determination
- Conclusions

What is the “Functional Containment?”

The collection of design selections that, taken together, ensure that:

- Radionuclides are retained within multiple barriers, with emphasis on retention at their source in the fuel, and
- Regulatory requirements and plant design goals for release of radionuclides are met at the Exclusion Area Boundary

HTGRs have Multiple Barriers to Radionuclide Release that Comprise the “Functional Containment”

- Fuel Kernel
 - Fuel Particle Coatings
 - Matrix/Graphite
- 
- Fuel Element
- Helium Pressure Boundary
 - Reactor Building

Modular HTGR Source Term Definition

- Quantities of radionuclides released from the reactor building to the environment during Licensing Basis Events. This includes timing, physical and chemical forms, and thermal energy of the release
- Modular HTGR Source Terms are:
 - Event-specific
 - Determined mechanistically using models of fission product generation and transport that account for reactor inherent and passive design features and the performance of the fission product release barriers that comprise the functional containment
 - Different from the LWR source term that is based on a severe core damage event

Modular HTGR Source Term Analysis

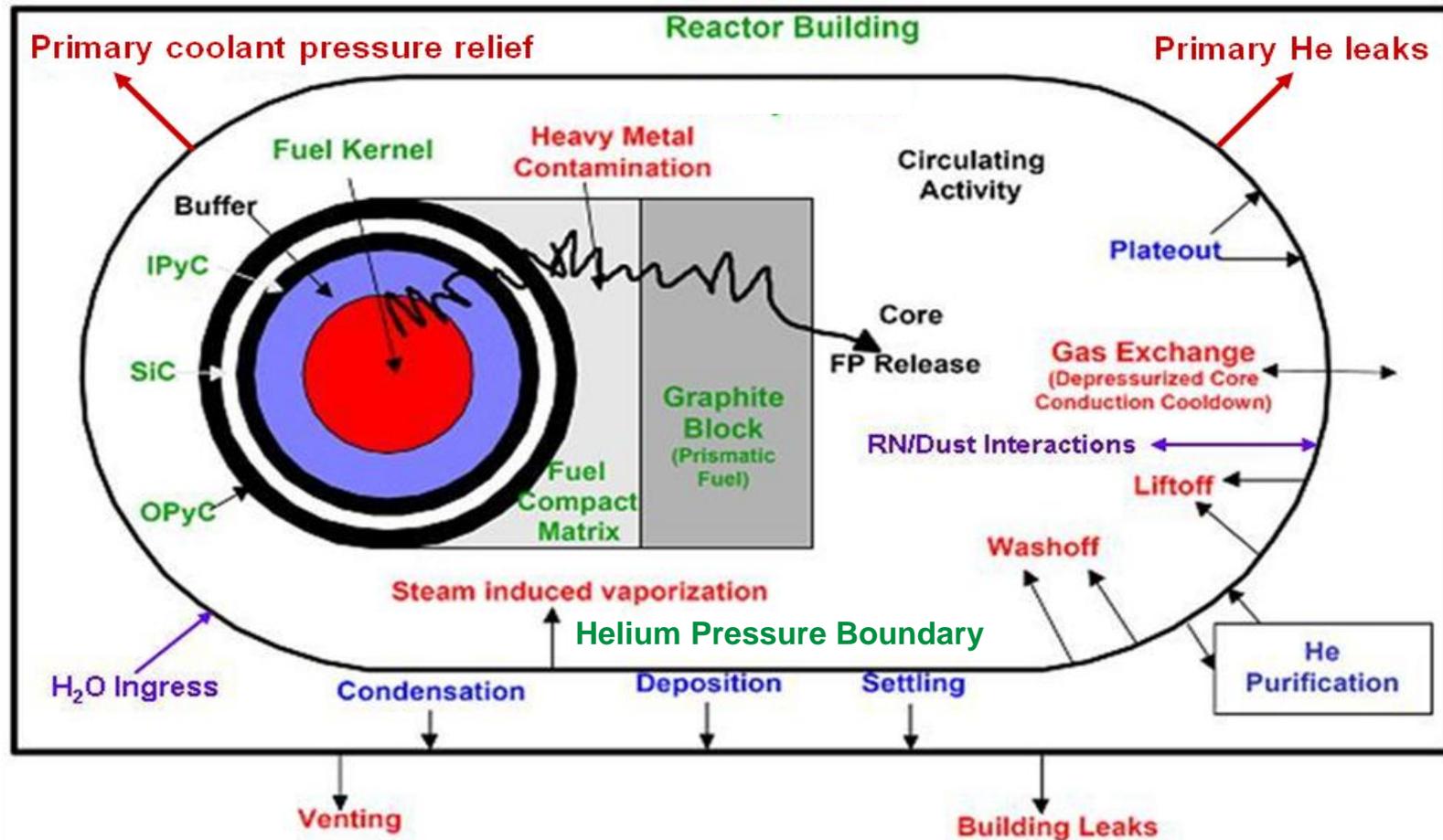
- Considers hundreds of radionuclides
- To facilitate analysis, fission products are grouped by chemical similarity and by similarity in transport properties
- Experience based on past analyses suggest that I-131, Cs-137, Cs-134 and Sr-90 are dominant contributors to offsite dose

Fission Product Transport Models Mechanistically Calculate

- Transport of radionuclides from their point of origin through the fuel to the circulating helium
- Circulating activity in the HPB
- Distribution of condensable radionuclides in the HPB
- Radionuclide release to and distribution in the reactor building
- Radionuclide release from the reactor building to the environment (source term)

In addition to providing source terms, these calculations provide radionuclide inventories throughout the facility that can be used for other purposes (shielding, worker dose, equipment EQ, etc.)

Modular HTGR Fission Product Retention



The phenomena illustrated in this figure are modeled to determine mechanistic source terms for normal and off-normal events

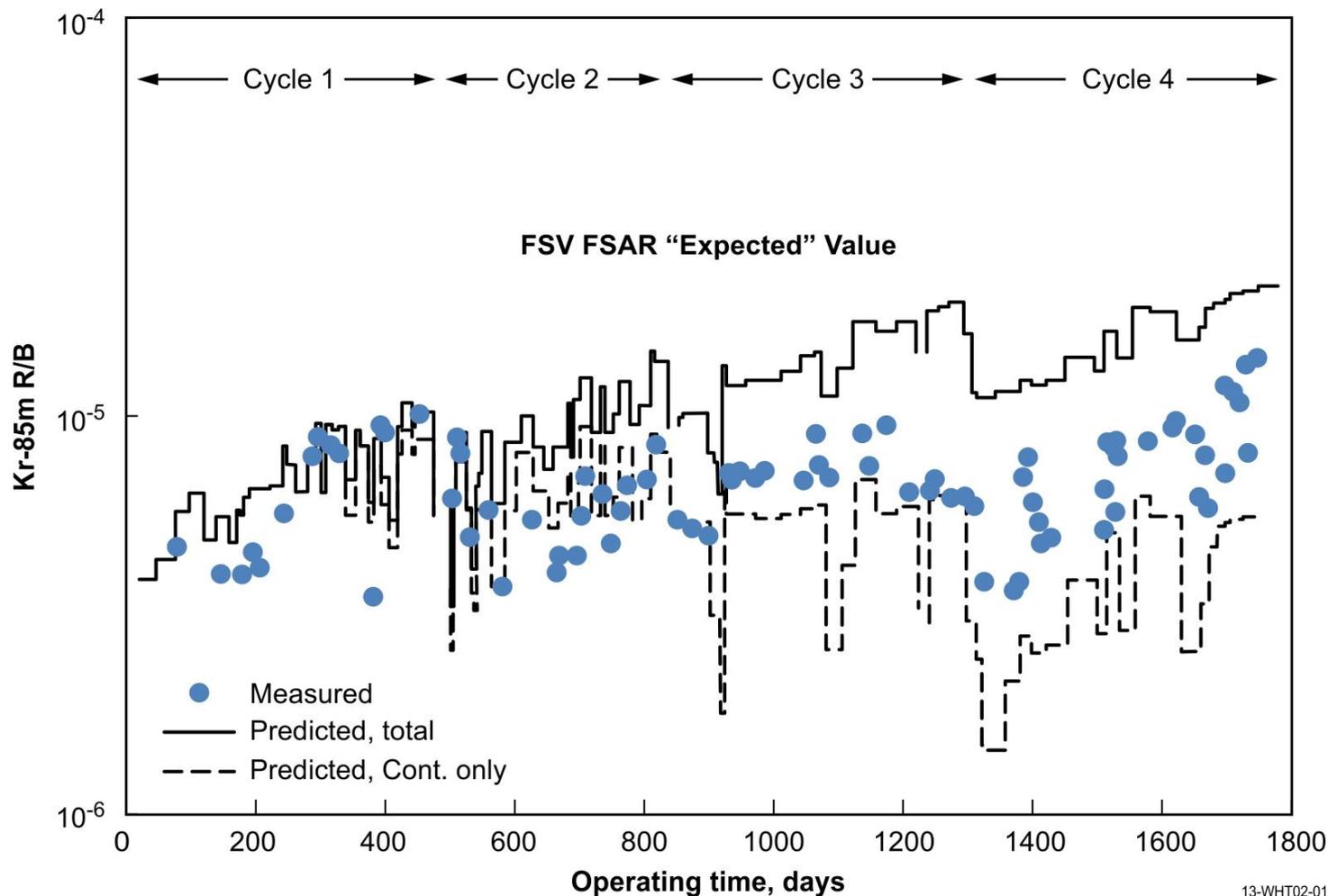
Fuel Particle Coatings are the Primary Barrier to Radionuclide Release During Normal Operation and Off-Normal Events

- Low heavy metal contamination and low initially defective fuel particles in as-manufactured fuel ($\sim 10^{-5}$)
- Minimal radionuclide release from incremental fuel failure during normal operation ($< 10^{-4}$)
- Minimal radionuclide release from incremental fuel failure during Licensing Basis Events ($< 10^{-4}$)
- Radionuclide release during LBEs dominated by exposed heavy metal (contamination and exposed fuel kernels)

Radionuclide Behavior During Normal Operation

- Most radionuclides reach a steady state concentration and distribution in the primary circuit (long lived isotopes like Cs-137 and Sr-90 are exceptions – plateout inventory builds up over plant life)
- Concentration and distribution are affected by:
 - Radionuclide half-life
 - Initial fuel quality
 - Incremental fuel failure during normal operation
 - Fission product fractional release from fuel kernel
 - Transport of fission products through particle coatings, matrix, and graphite
 - Fission product sorptivity on fuel matrix and graphite materials
 - Fission product sorptivity on primary circuit surfaces (plateout)
 - Helium purification system performance

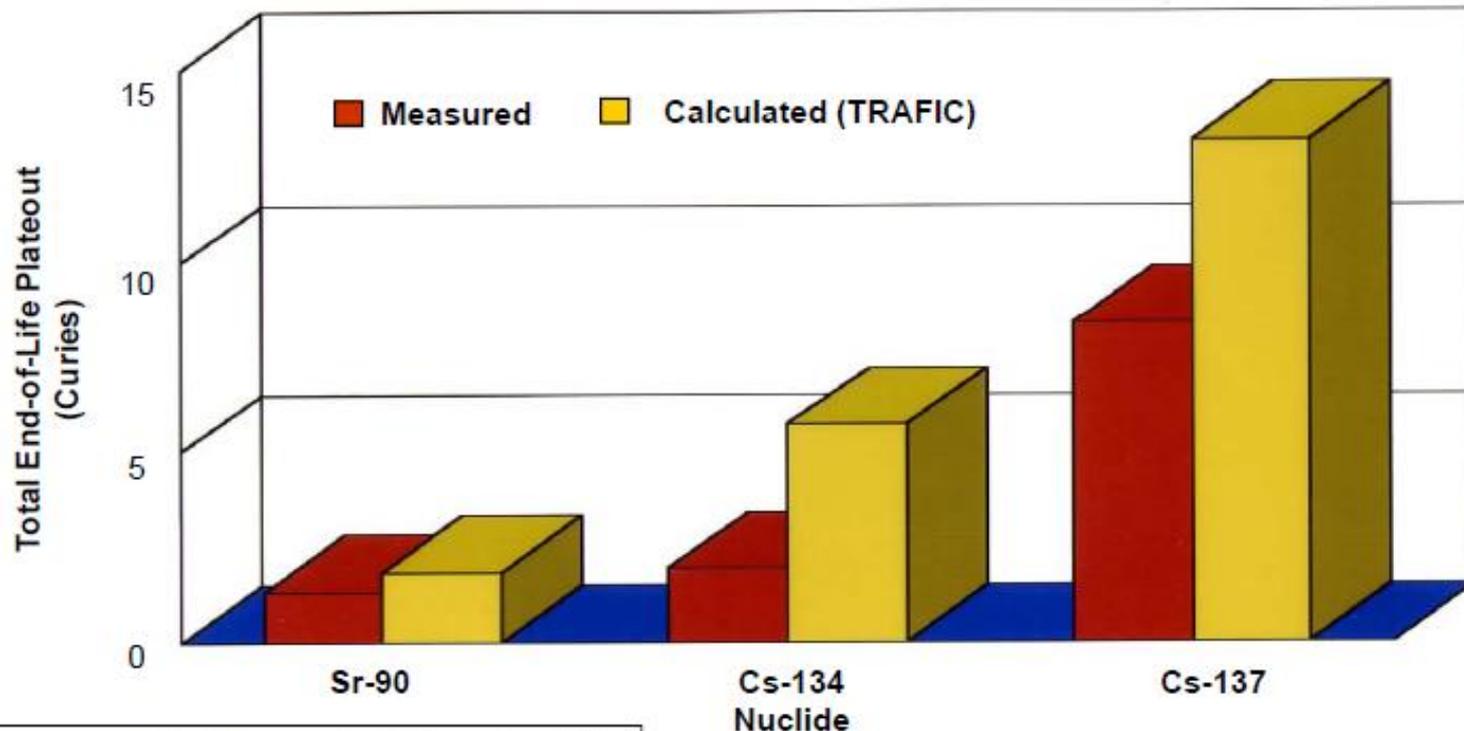
Comparisons of Calculated and Measured Fission Gas Release: Fort St. Vrain Kr-85m R/B – Normal Operation



Calculations using SURVEY code at General Atomics

13-WHT02-01

Comparisons of Calculated and Measured FSV Metallic Fission Product Release - Normal Operation



	Sr-90	Cs-134	Cs-137
Measured	1.3	1.9	8.4
Calculated	1.8	5.7	13.2

Calculations using TRAFIC Code at General Atomics

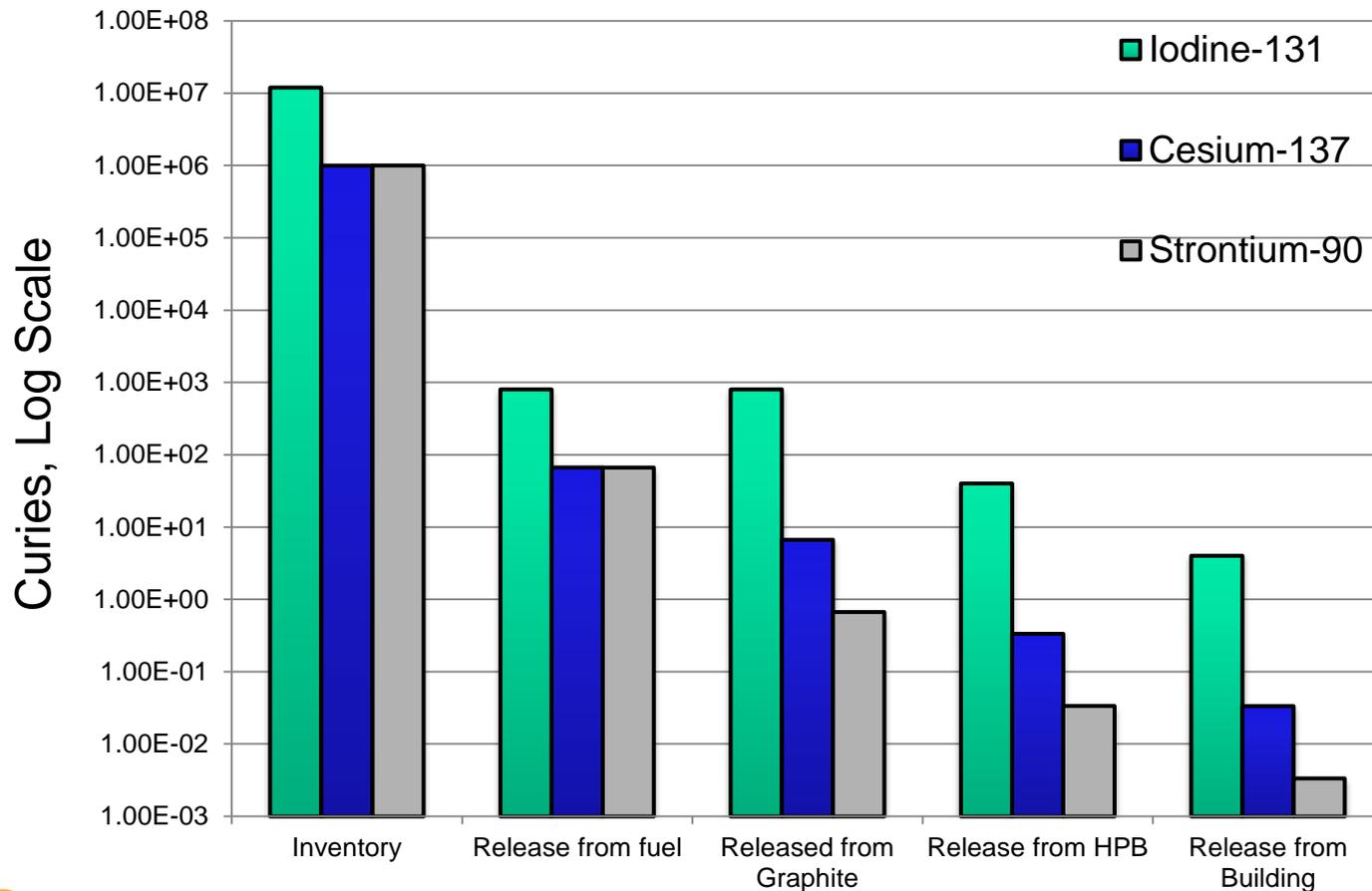
Early Release Mechanisms: Off-Normal Events

- Circulating activity
 - Released from HPB with helium in minutes to days as a result of HPB leak/break
 - Amount of release depends on location and any operator actions to isolate and/or intentionally depressurize
- Liftoff of plateout
 - For large breaks, fractional radionuclide amounts released from HPB with helium relatively quickly (minutes)
 - Amount of release depends on HPB break size and location. Surface shear forces must exceed those for normal operation to obtain liftoff
- HPB relief valve behavior
 - Sufficient moisture ingress can result in lifting of pressure vessel relief valve
 - Washoff of fractional radionuclide amounts can occur – can exceed liftoff fractions
 - Relief valve may cycle open/closed or may fail open

Delayed Release Mechanisms: Off-Normal Events

- Partial release from contamination, defective particles, particles failed in service, and particles that fail during off-normal events – tens of hours to days
- Delayed release from fuel is typically larger than circulating activity and any liftoff/washoff
- Amount of delayed release from fuel depends on time at temperature, level of oxidants, and radionuclide volatility
- Amount of delayed release from HPB depends on location and size of leak/break and on timing relative to expansion/contraction of gas mixture within the HPB
 - Small leaks have greater releases from HPB
 - Pressure relief valve behavior (reseating) affects release
 - Releases cease when temperatures within the HPB decrease due to core cooldown

Representative Functional Containment Performance During a Depressurized Loss of Forced Cooling*



Functional Containment Performance Summary

- Radionuclide retention within fuel during normal operation with relatively low inventory to HPB
- Limiting off-normal events characterized by
 - an initial release from the HPB depending on leak/break/pressure relief size
 - a larger, delayed release from the fuel
- Functional containment will meet 10CFR50.34 (10 CFR 52.79) at the EAB with margin for a wide spectrum of off-normal events without consideration of reactor building retention
- Functional containment (including reactor building) will meet EPA PAGs at the EAB with margin for wide spectrum of off-normal events

- Introduction
- Regulatory Background
- Functional Containment Performance and Mechanistic Source Term Determination
- ➔ • Conclusions

The NGNP Approach to Functional Containment and Mechanistic Source Terms

- Is consistent with the NRC Advanced Reactor Policy Statement
- Is consistent with discussions of containment function and mechanistic source terms in various NRC SECY documents and with approaches previously reviewed by the NRC staff for modular HTGRs
- Is event specific and can be applied to the full range of licensing basis events
- Uses mechanistic models of fission product generation and transport that account for reactor inherent and passive design features and the performance of the fission product release barriers that comprise the functional containment

Requested NRC Staff Positions – Recap

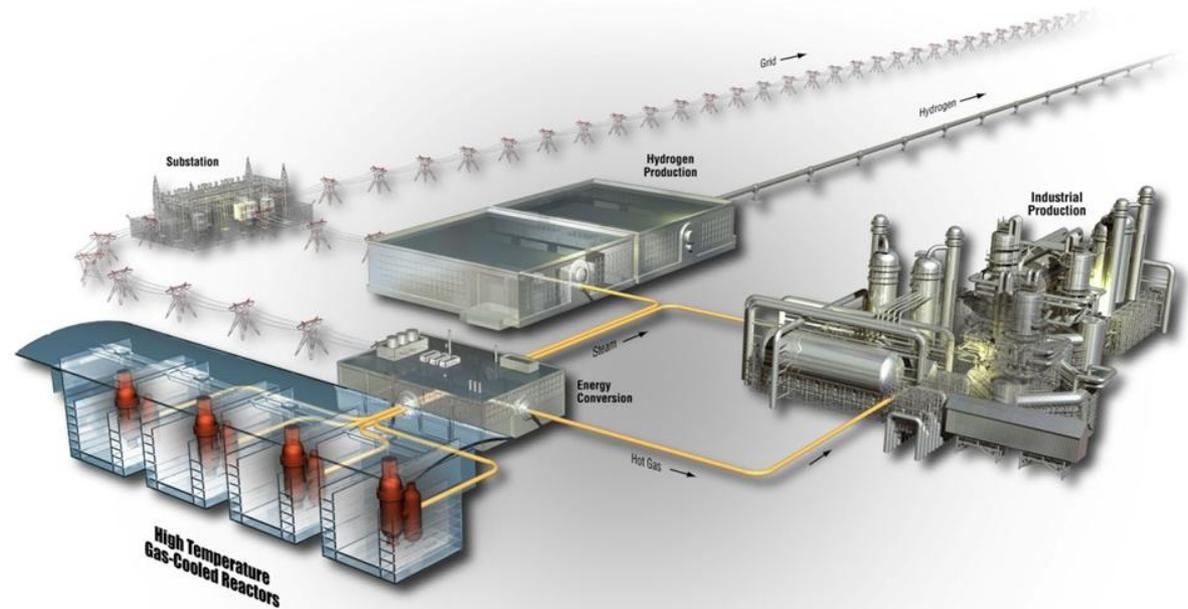
- Item 1.b: Establish options regarding functional containment performance standards
- Item 3.a: Endorse the proposed NGNP mechanistic source terms definition
- Item 3.b: Agree that NGNP source terms are event specific and determined mechanistically
- Item 3.c: Agree that NGNP has adequately identified the key HTGR fission product transport phenomena and has established acceptable plans for evaluating and characterizing those phenomena and associated uncertainties

Siting Source Terms

ACRS Future Plants Design Subcommittee Meeting

January 17, 2013

www.inl.gov



Meeting Agenda

- HTGR Safety Design Bases
- Licensing Basis Event (LBE) Selection Process
- Functional Containment Performance and Mechanistic Source Terms
- ➔ • Siting Source Terms
- Fuel Qualification and Radionuclide Retention

Siting Source Terms Presentation Outline

- Requested NRC Staff Position Regarding Siting Source Terms
- NGNP Siting Source Terms Approach
- Event Sequences Involving Graphite Oxidation
- SST Conclusions

- 
- Requested NRC Staff Position Regarding Siting Source Terms
 - NGNP Siting Source Terms Approach
 - Event Sequences Involving Graphite Oxidation
 - SST Conclusions

Requested NRC Staff Positions on Siting Source Terms (July 6, 2012 Letter)

- Item 1.c: Establish a staff position to support a final determination regarding how LBEs will be considered for the purpose of plant siting and functional containment design decisions, taking into consideration previous staff positions in SECY-95-299, that improved fuel performance is a justification for revising siting source terms and containment design requirements
 - In particular, we request that this staff position provide an adaptation of the guidance that has generally been applied to light water reactors (LWRs) for compliance with 10 CFR 100.21. (It is noted that for LWRs, this guidance has typically included the assumption of a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.)
 - The NRC's development of the NGNP adaptation of this guidance, which should reflect the NGNP's unique event response characteristics, will rely heavily on the establishment of the NRC staff positions associated with Licensing Basis Event Selection and establishing Mechanistic Source Terms

- Requested NRC Staff Position Regarding Siting Source Terms
- ➔ • NGNP Siting Source Terms Approach
- Event Sequences Involving Graphite Oxidation
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NGNP Approach to SSTs

- NGNP's approach to SSTs is patterned after that developed by DOE and the NRC staff in the development and review in the late 1980s and early 1990s of the MHTGR Conceptual Design documents including the PSID and PRA
 - Develop the design consistent with the safety design approach
 - Utilize risk insights as input to the design for the range of user and regulatory requirements
 - Select and mechanistically evaluate risk-informed LBEs including DBEs/DBAs as well as BDBEs, against the Top Level Regulatory Criteria (10CFR20, 10CFR50.34 and 52.79, and Prompt QHO) and the NGNP design goal (PAG at EAB)
- Consistent with MST approach, mechanistically evaluate events over LBE-spectrum that have limiting dose consequences for use as SSTs

MHTGR DBA Examples

- The MHTGR PSID identified several DBEs/DBAs and BDBEs enveloped by the following highest offsite consequence DBAs:
 - DBA-6: Steam Generator (SG) offset tube rupture with SG isolation and immediate and indefinite loss of forced cooling leading to an early (min to hr) and a delayed (days) radionuclide release from Helium Pressure Boundary (HPB) via opening of Vessel System (VS) relief valve to the Reactor Building (RB)
 - DBA-10: VS relief line breach of HPB with immediate and indefinite loss of forced cooling leading to an early (sec to min) and a delayed (days) radionuclide release from HPB to RB
 - DBA-11: Instrument line leak in HPB with immediate and indefinite loss of forced cooling leading to an early (min to hr) and a delayed (days) radionuclide release from HPB to RB
- Each of these DBAs entails ingress of moisture or air into the reactor

Bounding Event Sequences will also be Considered for Cliff Edge Effects

- To assure that there are no cliff edge effects and to understand the safety capability of HTGRs, supplement the LBE-derived SSTs with insights from a best estimate mechanistic evaluation of bounding event sequences, with the understanding that:
 - Such events shall be physically plausible rather than non-physical, arbitrary combinations of event parameters or end-states
 - While the bounding event sequences would not be rigorously quantified in terms of frequency, it is expected that they would generally have frequencies lower than the BDBE region
 - Events and their evaluation will consider the intrinsic and passive characteristics and the safety behavior of the HTGR

Process for Selection of Bounding Event Sequences

- Bounding event sequences will be selected based on a deterministic review of physically plausible events that potentially impact HTGR safety functions:
 - Remove core heat
 - Control core heat generation
 - Control chemical attack (e.g., graphite oxidation)
- The initial selection of bounding event sequences requires completion of preliminary design
- The bounding event selection process will use as a starting point the six MHTGR bounding event sequences requested by NRC staff in MHTGR PSID RAIs

MHTGR Bounding Event Sequences from NRC Staff

- BES-1 Inadvertent withdrawal of all control rods without scram for 36 hours (one module)
- BES-2 Station blackout (all modules) for 36 hours
- BES-3 Loss of forced cooling plus loss of RCCS for 36 hours (one module)
- BES-4 Steam generator tube rupture (25% of tubes) with failure to isolate or dump
- BES-5 Rapid depressurization (one module): double ended guillotine break of crossduct (sic) with failure to scram (assume RCCS failed for 36 hours and 25% unblocked thereafter)
- BES-6 External events consistent with those imposed on LWRs

Application of Bounding Event Sequence Analysis Results

- Analyses of bounding event sequences will be used to:
 - Identify and understand potential for “cliff edge effects” (i.e., high consequence events)
 - Determine potential risk significant plant or system vulnerabilities
 - Identify risk mitigation strategies as needed
- Analyses results will be documented as a part of the licensing application process

Previous NRC Staff Positions on MHTGR Bounding Event Sequences

- (1989) NUREG-1338
 - (p 15-7) – “The staff judges that these [bounding events proposed by the staff] results show that the MHTGR has the potential to cope with extremely rare and severe events without the release of a significant amount of fission products”
 - Appendix C, (p 4) – ACRS statement: “Neither the designers, the NRC staff, nor members of the ACRS have been able to postulate accident scenarios of reasonable credibility, for which an additional physical barrier to the release of fission products is required in order to provide adequate protection to the public”

- Requested NRC Staff Position Regarding Siting Source Terms
- NGNP Siting Source Terms Approach
- ➔ • Event Sequences Involving Graphite Oxidation
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Potential Event Sequences Involving Graphite Oxidation – Addressing SRM 93-092

- SRM 93-092 – “The Commission believes that, for the MHTGR, the staff should also address the following type of event. The loss of primary coolant pressure boundary integrity whereby air ingress could occur (from the "chimney effect") resulting in a graphite fire and the subsequent loss of integrity of the fuel particle coatings.”

Previous NRC Staff Positions for MHTGR Graphite Oxidation Event Sequences

- From (1989) NUREG-1338, Appendix B – Summary of BNL Independent Analysis in Support of Safety Evaluation Report, Section 3, Evaluation of Large Air Ingress Scenarios (p7) – “For the graphite oxidation to proceed to the point that structural damage inside the core would become possible, an unlimited air supply would have to be available for many days.”
- From (1995) NUREG-1338 (p 3-15) – “The staff concluded in draft NUREG-1338 that a graphite fire in the MHTGR core is a very low probability event. As stated in NUREG/CR-6218 on air ingress during severe accidents, without two breaches of the reactor vessel to create a chimney effect, it is not likely that significant amounts of air will enter into the core....Therefore, graphite fires are not a licensability issue for the MHTGR.”

NGNP Approach to Event Sequences Involving Graphite Oxidation

- Consistent with the findings of NUREG-1338 and the ACRS, it is expected that the frequency of the event type described in SRM 93-092 will fall so far below the LBE-spectrum of events (well below 5×10^{-7} per plant year) that the event would be considered incredible
- These expectations will be confirmed during the design process, once additional design detail is available
- Physically plausible bounding event sequences that maximize the potential for graphite oxidation will be considered in the bounding event sequence process as part of the NGNP licensing effort
- AGR Fuel Development and Qualification Program will obtain more data on air (and moisture) ingress effects

- Requested NRC Staff Position Regarding Siting Source Terms
- NGNP Siting Source Terms Approach
- Event Sequences Involving Graphite Oxidation
- ➔ • SST Conclusions

SST Conclusions

- The NGNP SSTs approach is essentially the same as that proposed by DOE in the MHTGR PSID and accepted by the NRC staff in NUREG-1338
- The approach is consistent with discussions of containment function and mechanistic source terms in more recent NRC SECY documents and with approaches previously reviewed by the NRC staff for modular HTGRs
- Limiting LBEs will be evaluated to determine SSTs
- Physically plausible Bounding Event Sequences, including those involving graphite oxidation, will be considered to ensure that there are no cliff edge effects

Requested NRC Staff Positions on Siting Source Terms – Recap

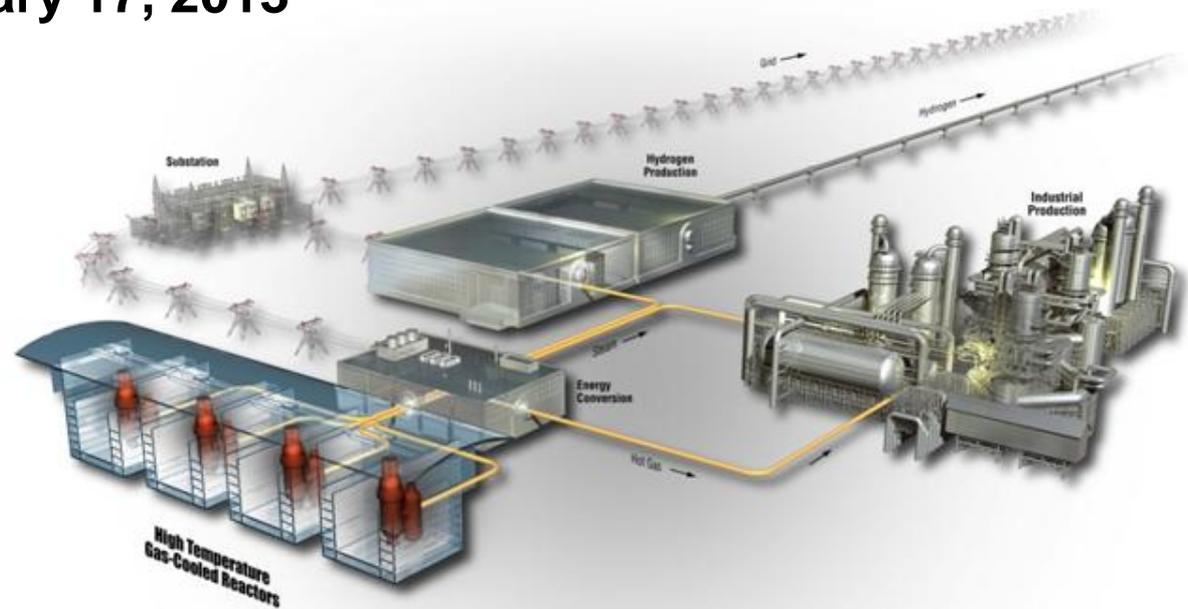
- Item 1.c. Establish a staff position to support a final determination regarding how LBEs will be considered for the purpose of plant siting and functional containment design decisions
 - In particular, we request that this staff position provide an adaptation of the guidance that has generally been applied to light water reactors (LWRs) for compliance with 10 CFR 100.21.

Fuel Qualification and Radionuclide Retention

ACRS Future Plant Designs Subcommittee Meeting

January 17, 2013

www.inl.gov



Meeting Agenda

- HTGR Safety Design Bases
- Licensing Basis Event (LBE) Selection Process
- Functional Containment Performance and Mechanistic Source Terms
- Siting Source Terms
- Fuel Qualification and Radionuclide Retention



Outline

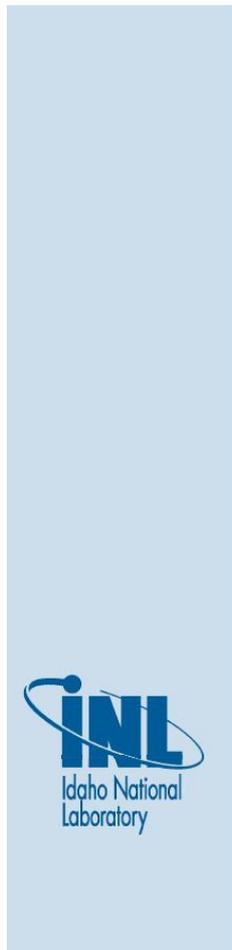
- NGNP White Paper and Requested NRC Staff Positions
- Background
- Fuel Qualification Approach
 - Key Questions
- Fuel Qualification Program: Plans, Status and Key Results as they relate to Licensing
 - Fabrication
 - Fuel Irradiation
 - Fuel Post-Irradiation Examination
 - Fuel Safety Testing
- Summary and Path Forward

White Paper

Next Generation Nuclear Plant Fuel Qualification White Paper INL/EXT-10-18610

July 2010

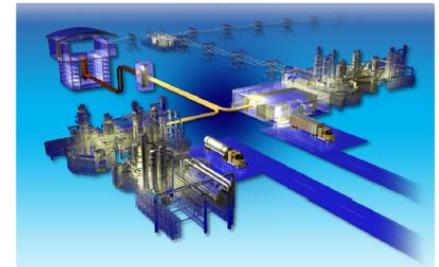
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Revision 0

NGNP Fuel Qualification White Paper

July 2010

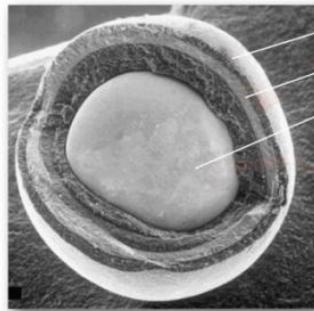


The INL is a U.S. Department of Energy National Laboratory operated by Battelle Energy Alliance.

July 6, 2012 NGNP Letter to Staff: Requested NRC Staff Positions on Fuel Qualification

Item 1.a: Confirm plans being implemented by the Advanced Gas Reactor Fuel Development and Qualification Program are generally acceptable and provide reasonable assurance of the capability of coated particle fuel to retain fission products in a controlled and predictable manner. Identify any additional information or testing needed to provide adequate assurance of this capability, if required.

TRISO Fuel



Pyrolytic Carbon
Silicon Carbide
Uranium Dioxide or Oxycarbide Kernel

Prismatic



Particles



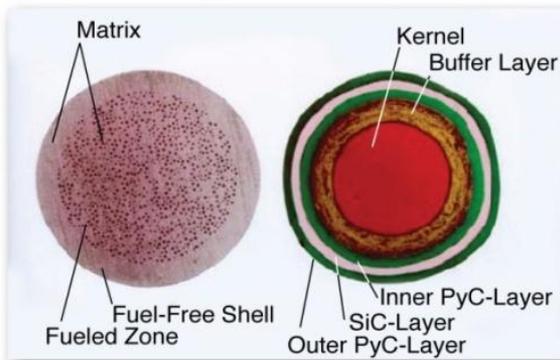
Compacts



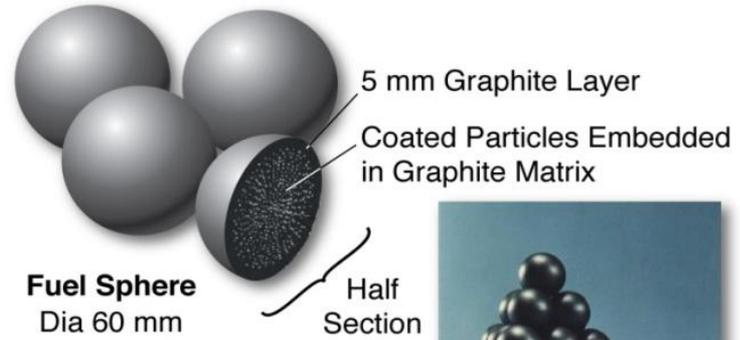
Fuel Element

TRISO-coated fuel particles (left) are formed into fuel compacts (center) and inserted into graphite fuel elements (right) for the prismatic reactor

Pebble



TRISO-coated fuel particles are formed into fuel spheres for pebble bed reactor



08-GA50711-01

Introduction

- Fuel's ability to retain fission products is extremely important to the safety case and licensing approach for modular HTGRs. Key principles for this fuel:
 - High quality, low defect TRISO fuel can be fabricated and characterized in a repeatable and consistent manner
 - Fuel performance with very low in-service failures is achievable within anticipated modular HTGR fuel design envelope and can be calculated to the requisite level of accuracy
- UCO is the fuel form being qualified
 - UCO a mixture of UO_2 , UC, and UC_2
 - Enables better fuel performance at higher burnup than UO_2 TRISO
 - UCO designed to provide excellent fuel performance at high burnup
 - Kernel migration suppressed (most important for prismatic designs because of larger thermal gradients)
 - Minimizes CO formation; internal gas pressure reduced
 - Fission products largely immobilized as oxides
 - Allows longer, more economical fuel cycle

Approach to NNGNP Fuel Qualification

- Establishment of a fuel product specification (kernels, coatings, compacts)
- Implementation of a fuel fabrication process capable of meeting the specification
- Implementation of statistical quality control procedures to demonstrate that the specification has been met
- Irradiation of statistically sufficient quantities of fuel with monitoring of in-pile performance and post-irradiation examination to demonstrate that normal operational performance requirements are met
- Safety testing of statistically sufficient quantities of fuel to demonstrate that accident condition performance requirements are met
- Data from the program are used to develop/improve and qualify models to predict fuel performance and fission product transport in the reactor

Key Questions to be Addressed in Fuel Qualification

- What are the reactor designer's quality and performance requirements for fuel?
- Can the fabrication process meet those requirements?
- Will UCO TRISO fuel be able to meet the performance requirements under normal operating conditions?
- Will UCO TRISO fuel be able to meet the performance requirements under accident conditions?
- How well do representative models predict what is being observed?
- What else have we learned about fuel behavior and fission product transport?

Answers based on results to date provided in red font in presentation

Nominal Maximum Service Conditions (Based on Historical MHTGR Designs)

<i>Parameter</i>	<i>Maximum Target Value</i>
Peak Fuel temperature – normal operation, °C	1,400
Maximum time averaged fuel temperature (normal conditions), °C	1,250
Peak Fuel temperature (accident conditions), °C	1,600
Fuel burnup, % FIMA	18 ^a
Fast fluence, 10 ²⁵ n/m ² (E > 0.18 MeV)	5
a. Estimated value for 15.5% enriched 425-μm reference fuel particle.	

Preliminary Fuel Quality and Performance Requirements (Based on Historical MHTGR Designs)

Parameter	NGNP – 750°C Core Outlet Temperature	
	Maximum Expected (Mean Value)	Design (95% confidence Value)
As-Manufactured Fuel Quality		
HM contamination	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Missing or defective buffer	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Missing or defective IPyC	$\leq 4.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
Defective SiC	$\leq 5.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
Missing or defective OPyC	0.01	0.02
In-Service Fuel Failure		
Normal operation	$\leq 5.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-4}$
Core heat-up accidents	$\leq 1.5 \times 10^{-4}$	$\leq 6.0 \times 10^{-4}$

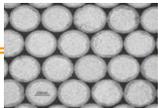
Key for source term analysis

Scaling Up Kernel Production, Coating, Overcoating and Compacting Processes to Create a Pilot Line

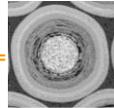
Lab Scale



Sol-Gel Kernel Production



Lab Scale 2 inch CVD Coating (60 g charge)



Prepare Matrix



Overcoat and Dry



Sieve



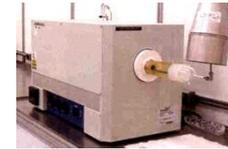
Table



Riffle



Compact



Carbonize



Heat Treat



Engineering Scale



Kernel Forming and Drying



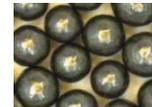
Industrial Scale 6 inch CVD Coating (2 kg charge)



Dry Mix and Jet Mill Matrix



Granurex Overcoat and Dry



Hot Press Compact



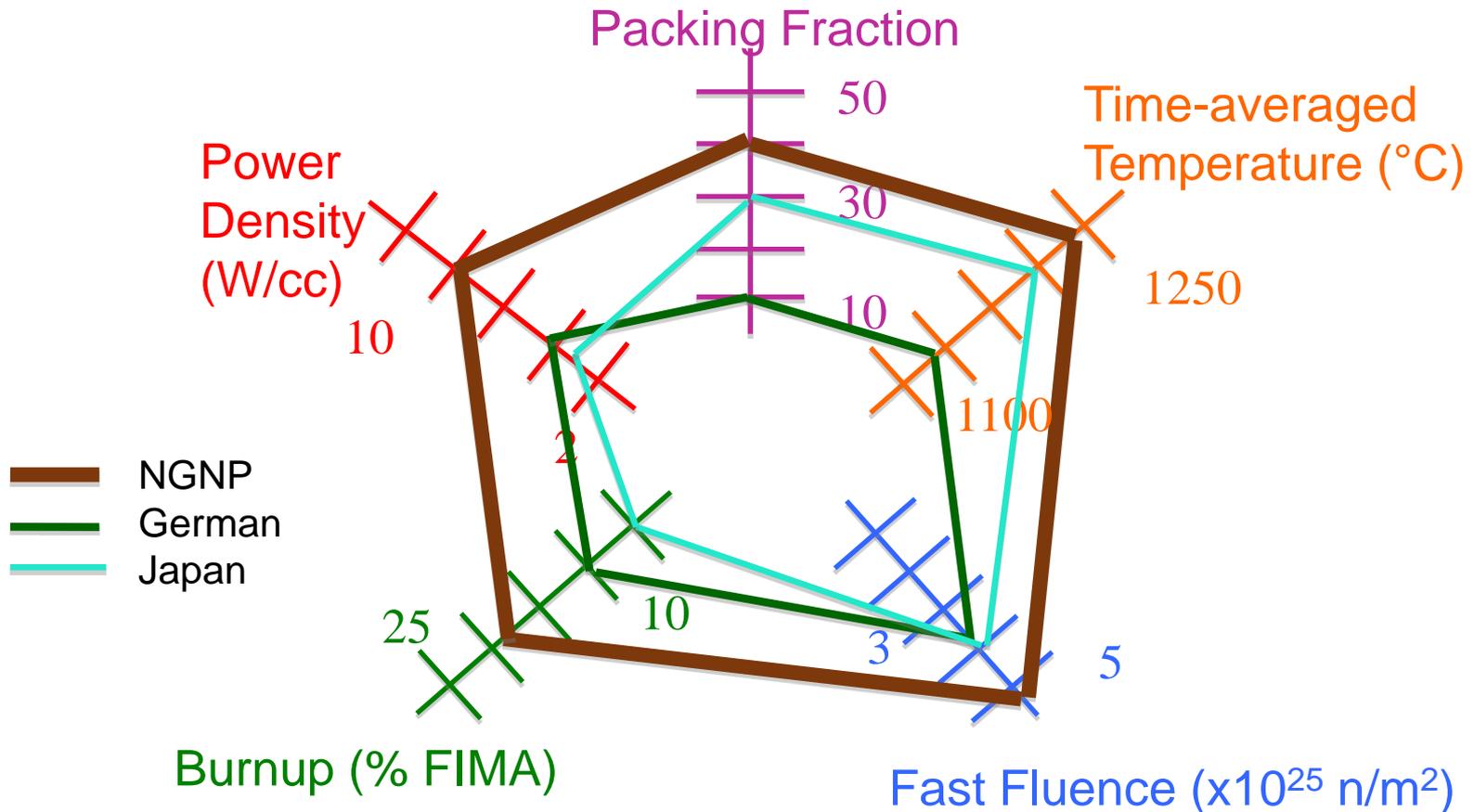
Carbonize + Heat Treat in one Sequential Process



Fuel Fabrication Accomplishments

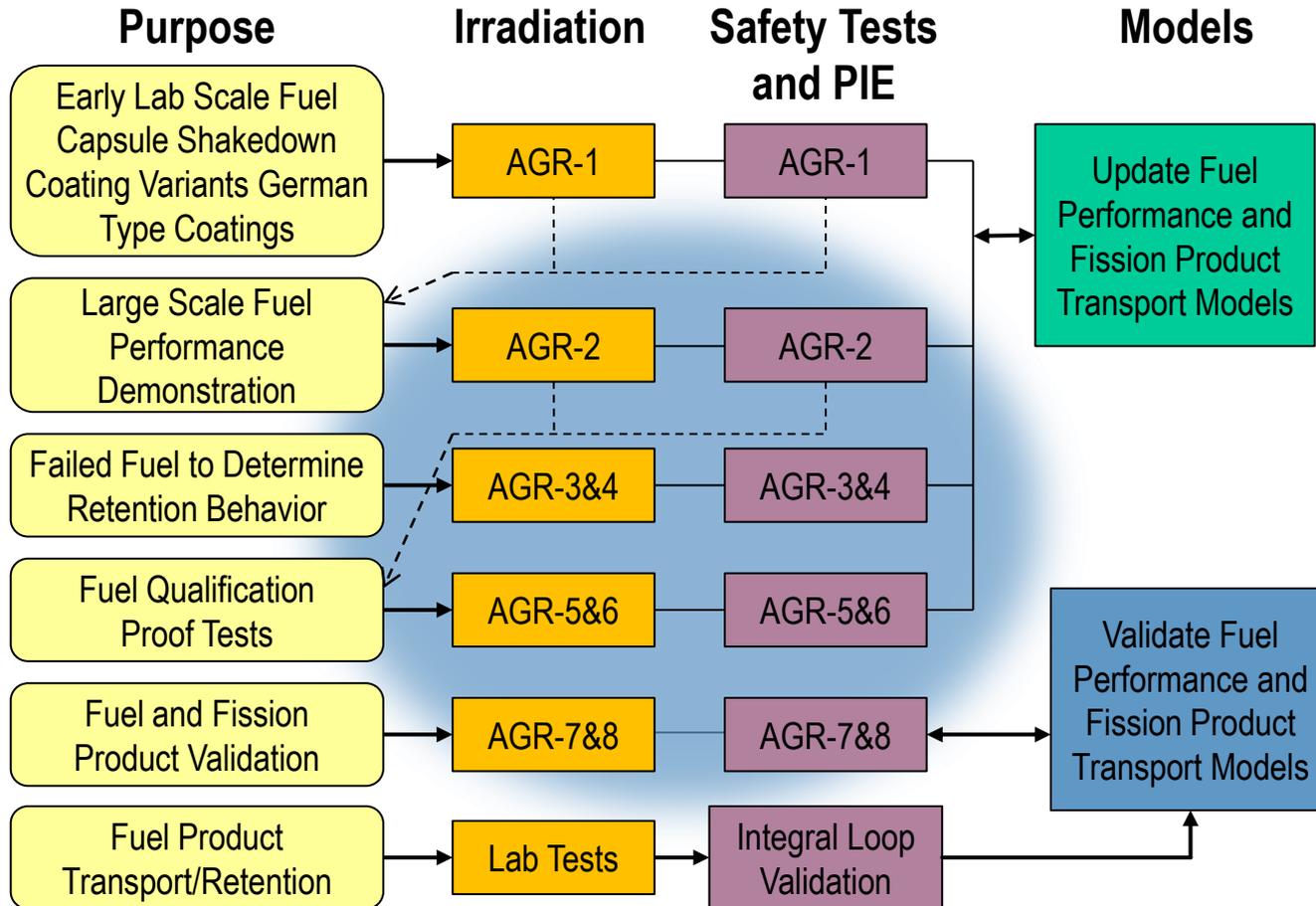
- Re-established capability to fabricate and characterize TRISO-coated particle fuel in the U.S. after a 10-15 year hiatus
- Developed a significantly improved understanding of how to fabricate high-performing TRISO fuel providing the technical basis for co-location of NGNP in industrial complexes
- Currently fabricating high-quality, low-defect TRISO-coated fuel particles in industry (B&W). **Can meet physical specifications and are almost meeting all defect specifications at 95% confidence. With larger sample sizes and a mature process, should meet the defect specifications in production mode**
- Vastly improved quality, reproducibility, process control, and characterization of TRISO fuel. Better control of the process, removal of high variability human interactions in the process, and better measurement technologies all contribute to better quality TRISO fuel
- Establishing a domestic vendor and associated fundamental understanding of key fuel fabrication parameters establishes credibility that the historical industrial experience from Germany in the 1980s is repeatable and has a sound technical basis
- All technologies needed to establish a pilot line are in industrial hands. Qualification fuel for AGR-5/6/7 will be produced in 2013

Performance Envelope for NGNP TRISO Fuel is more Aggressive than previous German and Japanese Fuel Qualification Efforts



Radar plot of five key parameters of fuel performance

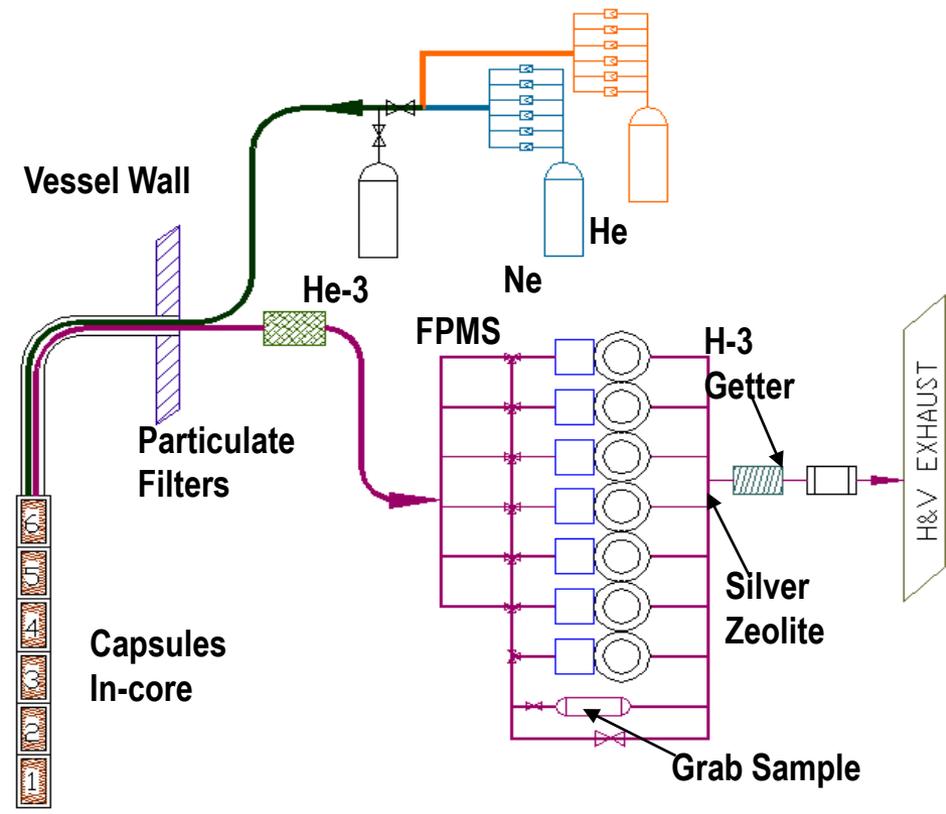
Overview of AGR Program Activities



Moisture and air ingress effects are part of AGR-5/6 PIE

NGNP Fuel Irradiation Capsule AGR-1 Demonstrated Outstanding Performance

- 350 um UCO TRISO; 19.7% enriched
- Goal burnup ~18–19% FIMA
- $\langle T \rangle_{\max} < 1250^{\circ}\text{C}$, $\langle T \rangle_{\text{avg}} \sim 1150^{\circ}\text{C}$
- Fast fluence $< 5 \times 10^{25} \text{ n/m}^2$
- Irradiation began in December 2006 and completed November 2009
- *Peak burnup of 19% FIMA with no failures out of 300,000 particles*



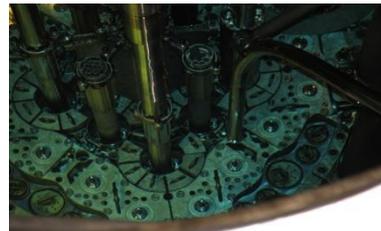
Individual capsule assembly with fuel



Completed test train



Insertion into INL ATR

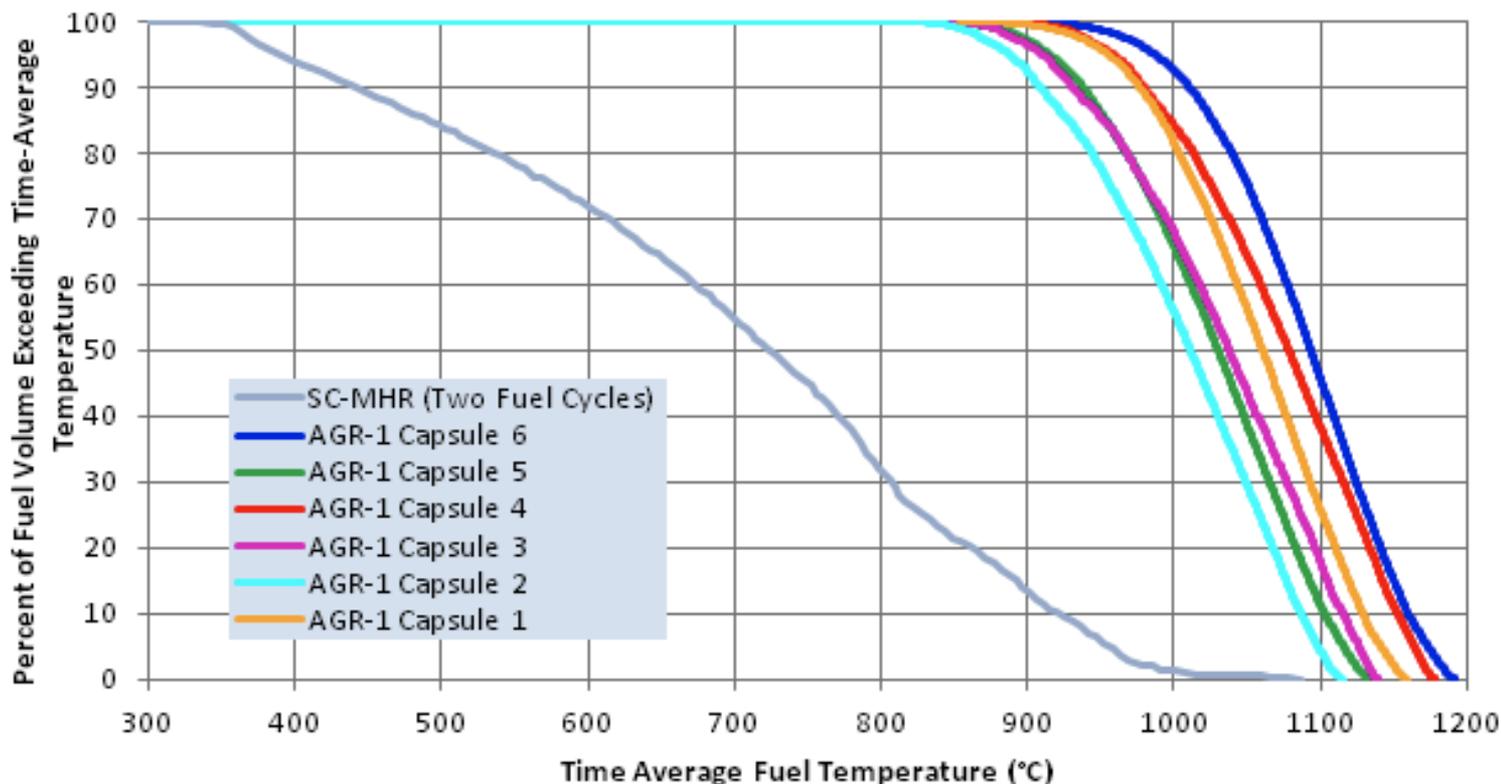


FPMS system



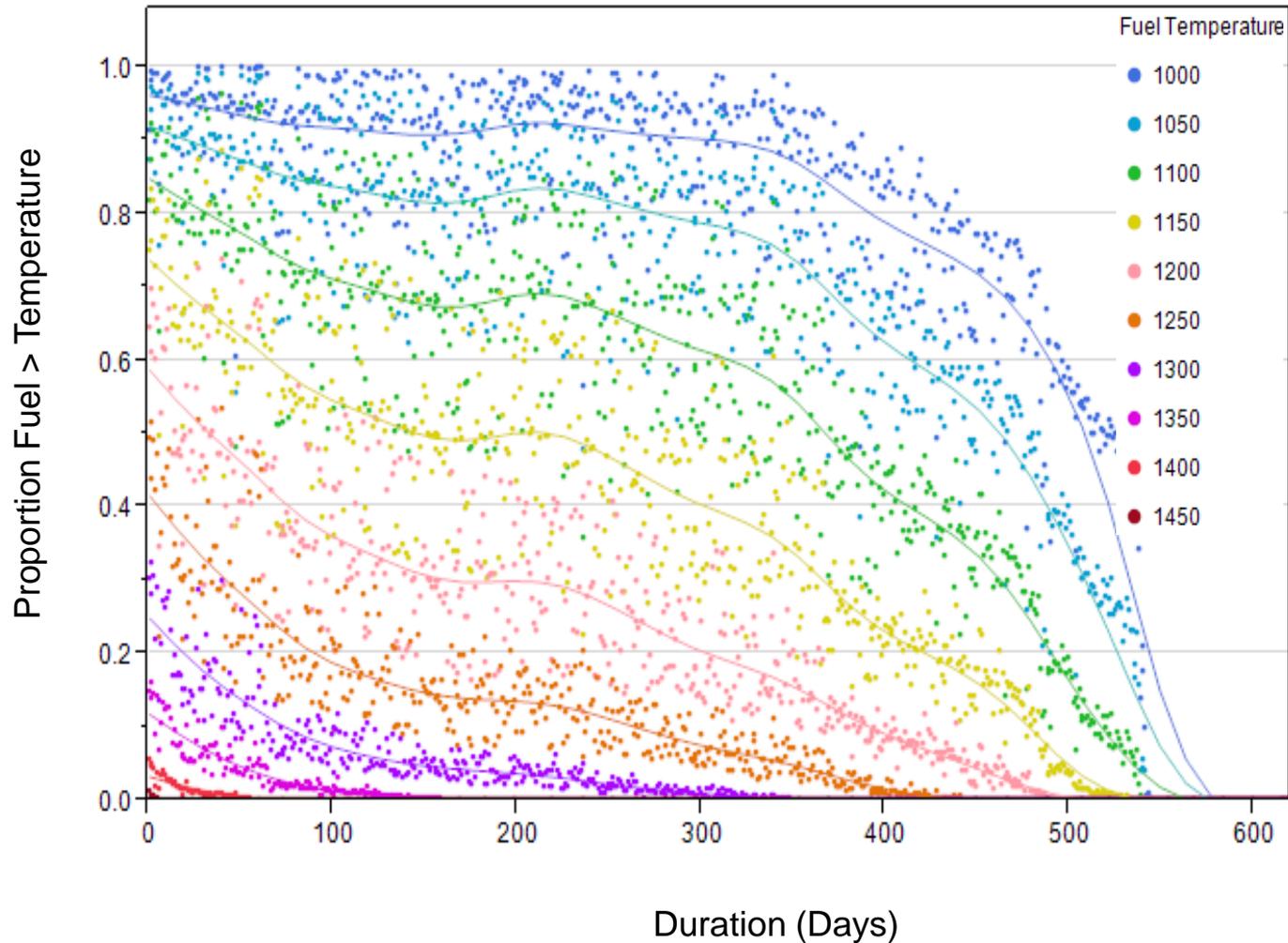
A wide range of temperatures, burnups and fluences were experienced by AGR-1 fuel compacts. Temperatures bound that expected in reactor.

**Time-Average Fuel Temperature Distribution
AGR-1 vs SC-MHR**



SC-MHR is General Atomics conceptual design for NGNP

Large Quantities of TRISO Fuel Particles in the AGR-1 Irradiation Spent Significant Time at High Temperature



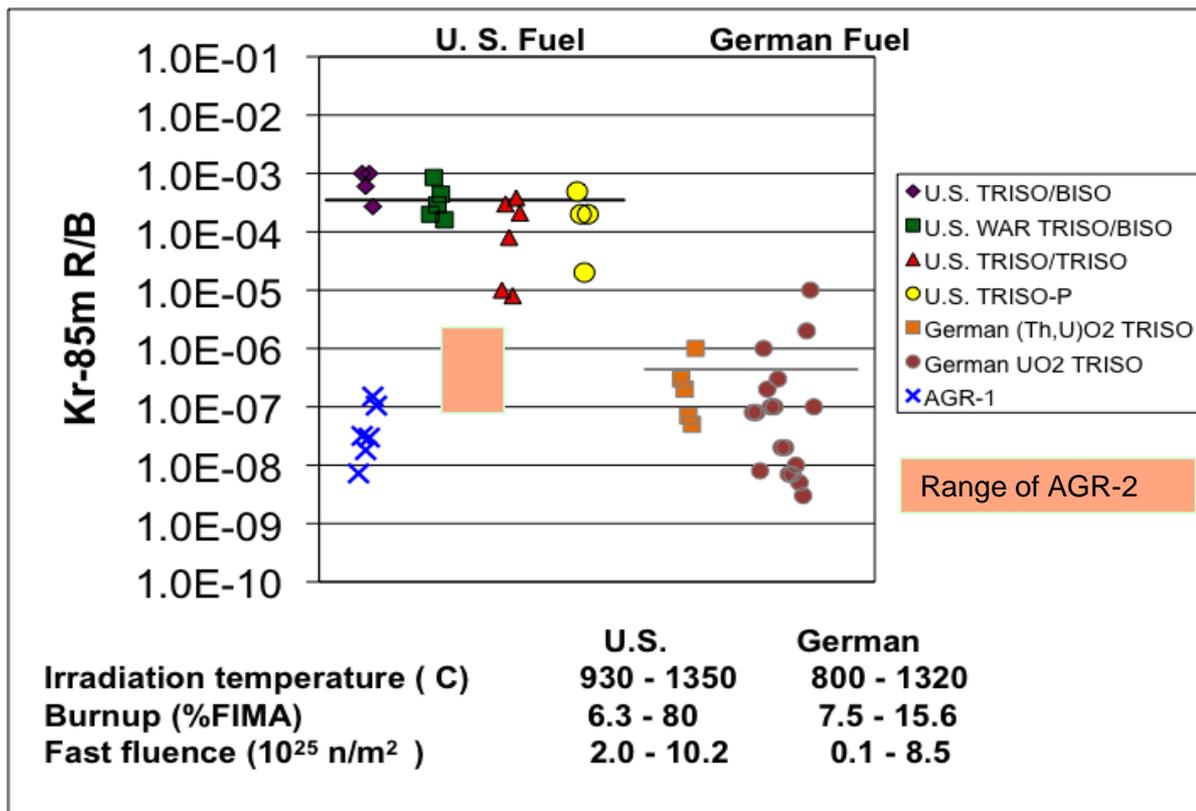
AGR-2 is Testing Vendor-produced UO₂ and UCO TRISO Fuel

Capsule	Fuel Type	Vendor	Enrichment	Peak Burnup Goal	Time-Average Peak Temperature
6	425 μm UCO	B&W	14%	12% FIMA	<1250°C
5	425 μm UCO	B&W	14%	14% FIMA	<1250°C
4	500 μm UO ₂	PBMR	9.6%	11% FIMA	<1150°C
3	500 μm UO ₂	B&W	9.6%	11% FIMA	<1150°C
2	425 μm UCO	B&W	14%	14% FIMA	<1400°C
1	500 μm UO ₂	CEA	19.6%	16% FIMA	<1150°C

Irradiation began in June 2010. Expected to complete in September 2013

AGR-1 and AGR-2 TRISO Fuel R/B Results Demonstrate Excellent Fuel Performance

Release-to-birth ratio (R/B) is measure of gas release from the fuel and a direct indicator of fuel performance

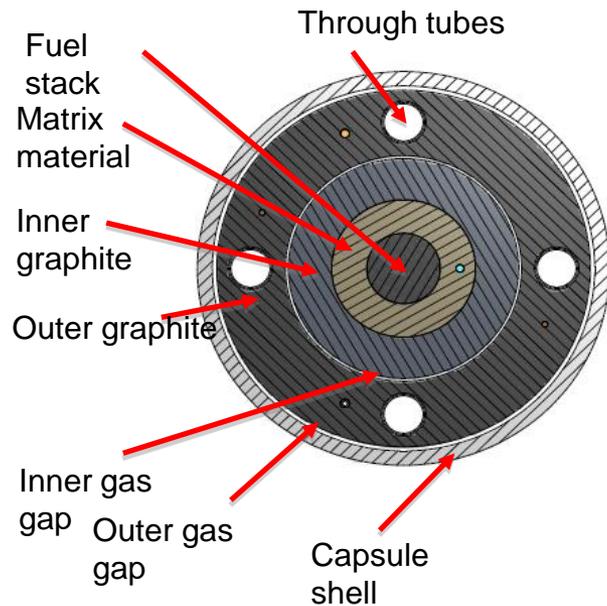


Fission Product Transport: Supporting a Mechanistic Source Term

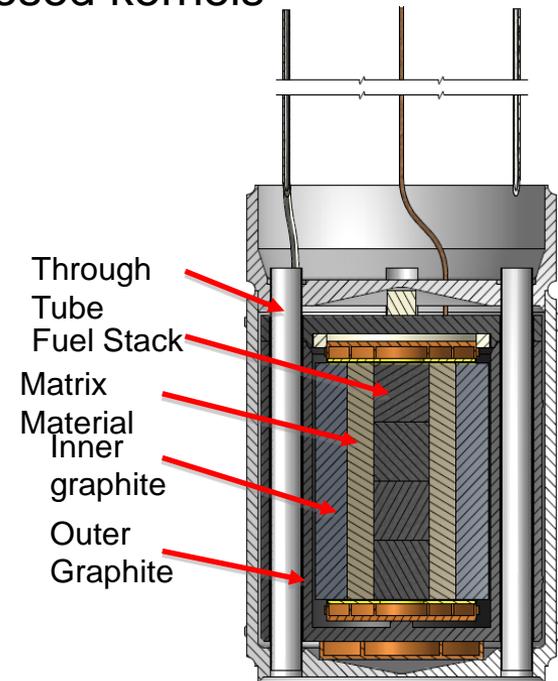
- NGNP will use a mechanistic source term that takes into account the performance of all fission product release barriers (kernels, coatings, compact matrix, graphite, helium pressure boundary, reactor building) to meet radionuclide control requirements
- Goal is to provide technical basis for mechanistic source terms under normal and accident conditions to support reactor design and licensing
- Experimental data to be generated by irradiation experiments (AGR-3/4 and 8), PIE, safety testing, and loop testing
- Independent validation experiments are part of the plan

AGR-3/4 has Designed-to-fail Fuel and is Performing as Expected

- Need to understand the behavior of fission products released from the small fraction ($\sim 10^{-5}$) of defective fuel and retained in graphitic components in the core
- Use “designed-to-fail” fuel that will provide a known source of fission product release
- Determine release rates of radionuclides from exposed kernels
- Establish metallic fission product transport and retention in fuel matrix and fuel element graphite
- Twelve separate capsules to span the temperature, burnup, and fluence envelope



AGR-3/4 Capsule Cross Section



Axial Cutaway of One of the 12 Capsules

TRISO Fuel Irradiation Qualification Accomplishments

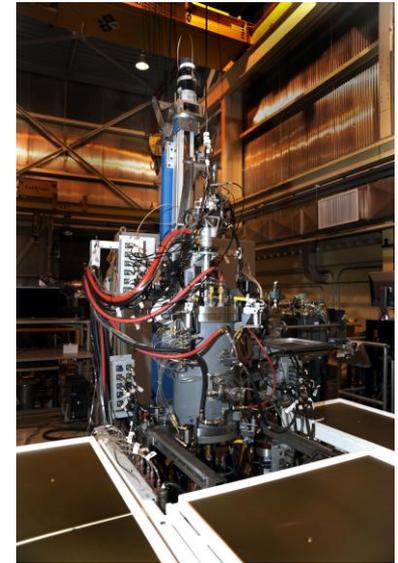
- Completed most successful U.S. irradiation of TRISO-coated particle fuel (AGR-1). 300,000 particles tested to peak burnup of 19.4% FIMA, a peak fast fuel of 4.5×10^{25} n/m² and peak time-average peak temperatures of 1250°C (peak MHTGR service conditions) with no failures
 - The expected superior irradiation performance of UCO at high burnup has been confirmed - no kernel migration, no evidence of CO attack of SiC, and no indication of SiC attack by lanthanides
- The AGR-1 95% confidence failure fraction is $<1E-5$, a factor of 20 better than the design in-service failure fraction of $2E-4$. The more severe AGR-1 irradiation conditions compared to the vast majority of historic modular HTGR designs suggest substantial fuel performance margin
- Irradiation of AGR-2 is underway; no failures to date. Completion in September 2013
- Irradiation of AGR-3/4 is underway to study release/retention of fission products from failed TRISO fuel. Will complete in April 2014. This experiment will provide data needed for source term evaluations for UCO TRISO fuel, new matrix and graphite

Post-Irradiation Examination Activities

- Infrastructure to meet fuel objectives is largely in place
- Objectives:
 - Detailed characterization of fuel behavior after irradiation in the reactor
 - Mass balance of fission products is critical for reactor source term
 - High temperature safety testing is required to establish fuel behavior under accident conditions

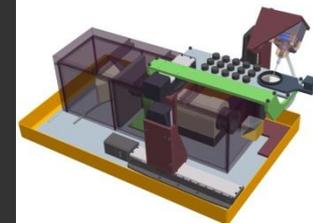
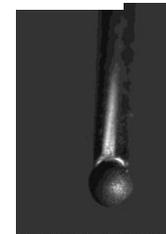
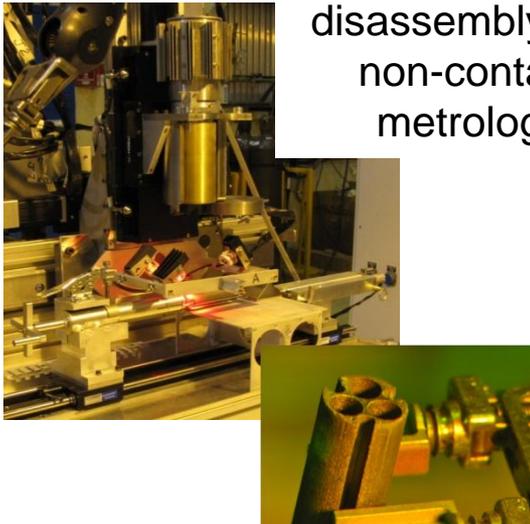


ORNL Furnace

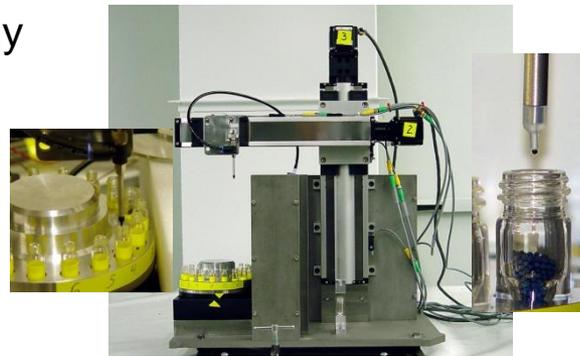


INL Furnace

Capsule
disassembly and
non-contact
metrology

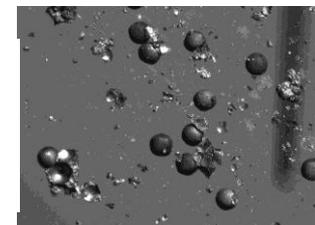


Particle handling and inspection



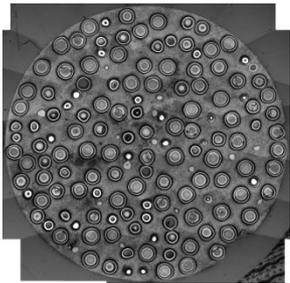
Advanced-IMGA

Deconsolidated
AGR-1 particles
and matrix



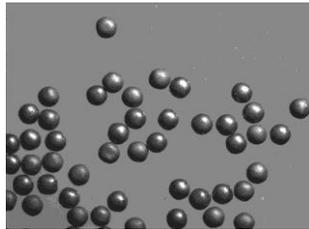
Using advanced characterization techniques to characterize fuel and fission product interactions from the millimeter to nanometer scale is improving our understanding of TRISO fuel behavior

10 mm



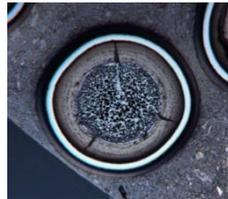
Cross-section of Compact

500 μm



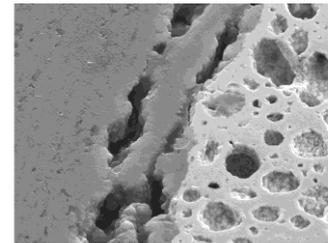
Individual Particles

100 μm



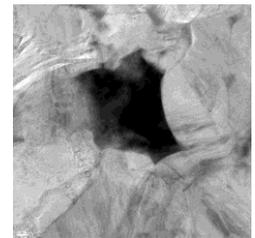
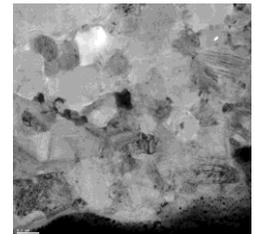
Micrograph of Particles

1 μm



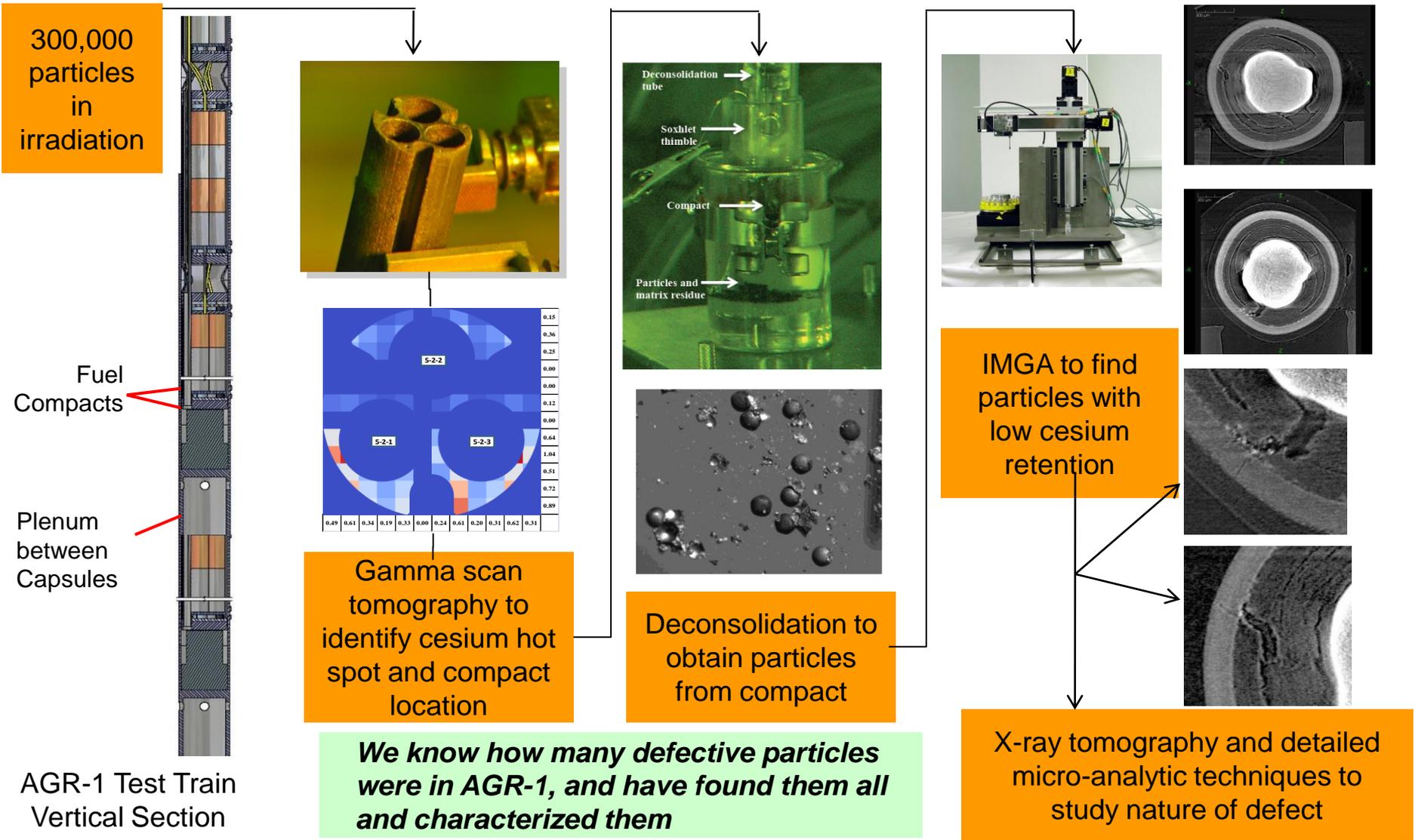
SEM Image of Buffer Kernel Interface in Particles

1 nm



High Resolution FIB/TEM Images of Precipitate near IPyC/SiC Interface

Methodology for isolating and identifying failed TRISO fuel particles greatly improves ability to characterize and understand fuel performance



NRC Follow-up Item: ATR irradiations do not produce enough Pu (and thus Pd) relative to that in the reactor and Pd corrosion of SiC at high burnup in TRISO fuel is an important degradation mechanism

- Much more Pd produced from Pu fissions than U fissions. Important in high burnup LEU fuel
 - Concentration of Ag and Pd produced in AGR-1 during ATR irradiation is about 40 and 33% respectively below that expected in a prismatic HTGR at a peak burnup of ~20% FIMA
 - FTE-13, a test of PuO₂ TRISO fuel (Peach Bottom), was irradiated to 70% FIMA and typical HTGR temperatures and levels of radiation damage. Some Pd interaction with SiC was observed, but no large-scale degradation of the SiC layer was observed.
 - Volumetric Pd concentration in FTE-13 with PuO₂ kernels is 75x that of AGR-1. Areal concentration is 60x. Concerns raised by NRC about Pd attack are not expected to be significant under NGNP irradiation conditions
- The concentration differences in AGR irradiations and the HTGR are small compared to the level of Pd generated in FTE-13. AGR-1 PIE is providing new understanding of Pd interactions with SiC that suggests Pd is less of an issue than previously thought

NRC Follow-up item: Will high Ag or Pd release cause degradation of SiC and allow Cs release from the particles? Is there an enhancement to cesium diffusion under irradiation?

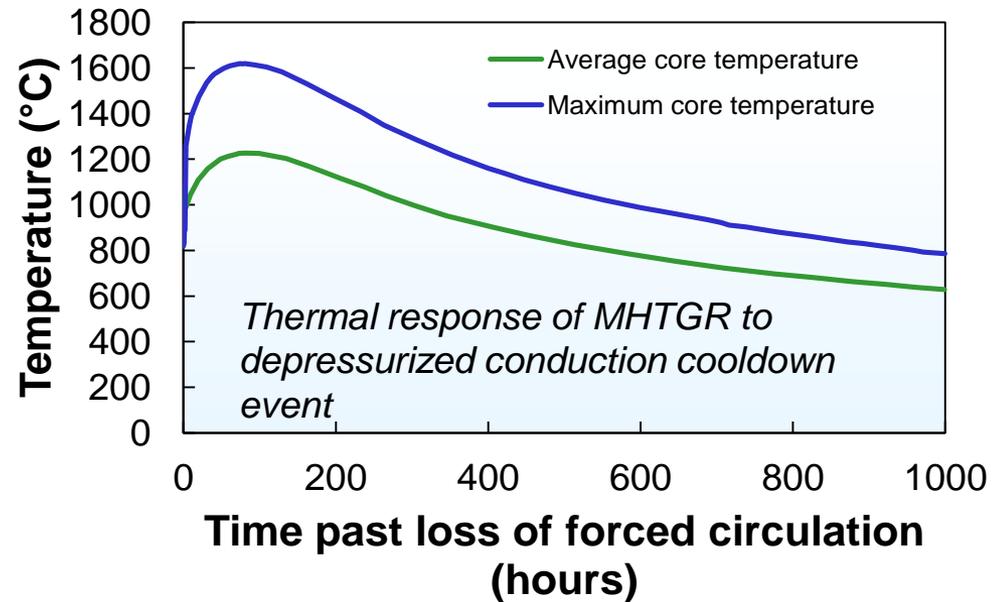
- No evidence of Ag and/or Pd affecting Cs release under normal or accident conditions in German database
 - No evidence that Ag or Pd release affects Cs release from AGR-1
 - Large Ag release in AGR-1, but no cesium release
 - ~ 1% of Pd is outside the SiC in AGR-1, but SiC layer is retentive of Cs and no “attack” has been observed
 - Minimal release of Cs to the matrix implies no substantial Pd degradation of SiC layer
 - No release of Cs in AGR heating tests (at 1600 and 1700°C) to date unless compact had an SiC defect
 - Minimal release of Cs to the matrix implies no enhanced diffusion under irradiation. Low releases to the matrix suggest IAEA TECDOC diffusion coefficient at normal operating temperatures is conservative
- AGR-1 data show no evidence of degradation or enhancements effects influencing cesium release/diffusion

TRISO Fuel Post-Irradiation Examination (PIE) Accomplishments

- Post-Irradiation examination is revealing new understanding of fuel performance and fission product transport
 - Characterization of kernel and coating behaviors to better understand performance and potential failure modes
 - More complete mass balance of key fission products (Ag, Cs, Sr, Eu, Ce, Pd)
 - IMGA to examine particle to particle variability and to identify defective particles that release fission products
 - Gamma scanning of test train components and deconsolidation of fuel compacts to evaluate retentiveness of SiC layer
 - Fission product/SiC interactions
 - Characterizing fuel and coating layer microstructures at micro and nano-scale
 - No Pd corrosion or attack of SiC has been observed! **Models overpredict SiC corrosion by Pd**
 - **Models conservatively overpredict release of Cs from fuel under normal operation**

Accident Safety Testing of TRISO Fuel

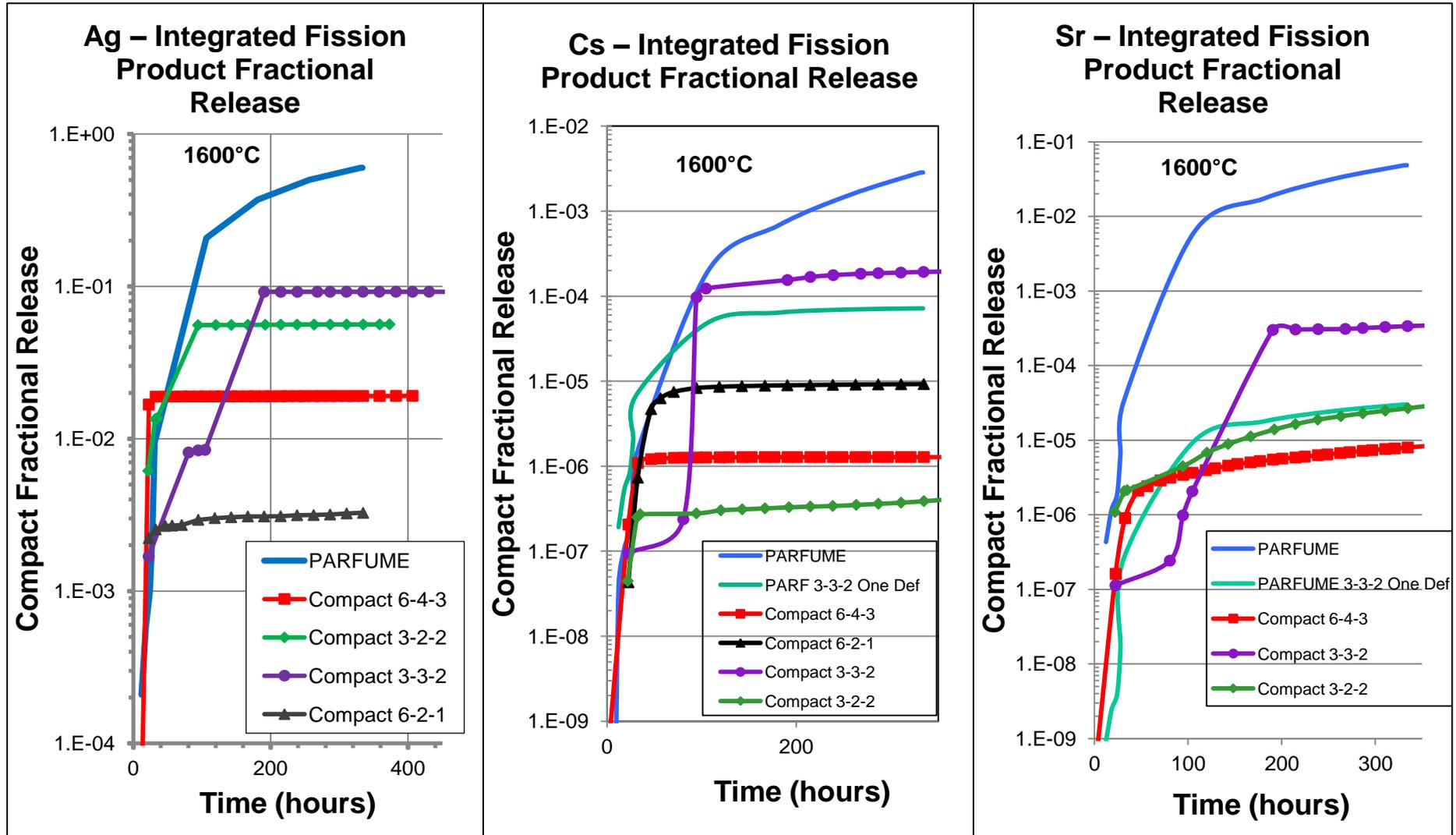
- Simulate heatup of fuel compacts following depressurized conduction cooldown event
- Isothermal testing for hundreds of hours at 1600, 1700, and 1800°C
- Five isothermal 1600 and 1700°C tests have been completed
- An 1800°C isothermal test will be performed this year
- Actual time-temperature test to be performed this year
- Testing of deconsolidated particles will occur in late 2013 or early 2014



Key Results

- Releases are very low unless a defective particle is present
- Releases are from fission products that diffused into the matrix during the irradiation and not from the intact TRISO particles during the high temperature heating

Historical IAEA TECDOC data used in models significantly overpredicts release under AGR isothermal heating tests



TRISO Fuel Safety Testing Accomplishments

- Accident safety testing of UCO TRISO from AGR-1 is nearing completion and demonstrating robustness of fuel
 - Very low releases after hundreds of hours at 1600 and 1700°C. No particle failures (no noble gas release measured)
 - Releases are associated with fission products that diffused into the matrix during the irradiation. No diffusive release from intact TRISO particles during the high temperature heating
 - UCO TRISO fuel should be able meet in-service failure fraction under off-normal conditions, but more data are needed to demonstrate statistical significance. Still need data on performance under water and air ingress conditions
 - Historical database of diffusion coefficients significantly overpredicts measured releases

Path Forward

- Complete safety testing and PIE on AGR-1 fuel in 2013 (including safety testing at 1800°C)
- Complete AGR-2 performance demonstration irradiation in 2013
- Complete AGR-3/4 source term irradiation in 2014
- Perform PIE and safety testing of AGR-2 and AGR-3/4 in 2014-2015
- Fuel qualification and margin irradiation (AGR-5/6/7) is scheduled for 2016
- Moisture and air effects on fuel are scheduled as part of PIE campaign for AGR-5/6/7 in 2018-2020
- AGR-8 will follow AGR-5/6/7

Key Results of On-going Research

Fabrication

- Improved understanding of TRISO fuel fabrication process
- Improved fabrication and characterization of TRISO fuel produced by fuel vendor

Irradiation Performance

- Outstanding irradiation performance of a large statistically significant population of TRISO fuel particles under high burnup, high temperature HTGR conditions
- Expected superior irradiation performance of UCO at high burnup has been confirmed

Post-Irradiation Examination and Safety Testing

- Post-Irradiation examination of AGR-1 indicates:
 - Ag release consistent with model predictions
 - No Cs release from intact particles under irradiation
 - No Pd attack or corrosion of SiC despite large amounts of Pd outside SiC
- Initial safety testing at 1600 and 1700°C demonstrating robustness of UCO TRISO under depressurized conduction cooldown conditions
 - Low releases from intact particles. Releases attributed to defective particles and transport of fission products released during irradiation. No particle failures observed to date

Summary

- The AGR Fuel Development and Qualification Program will provide data necessary to better understand fuel performance and fission product behavior for modular HTGRs
- The AGR Fuel Program is laying the technical foundation needed to qualify UCO TRISO fuel made to fuel process and product specifications within an envelope of operating and accident conditions that are expected to be bounding for modular HTGRs
- AGR results to date are consistent with current design assumptions about fuel performance and radionuclide retention. Program is obtaining additional data to support model development and validation
- AGR results to date are consistent with the safety design basis, including the functional containment and mechanistic source term approaches presented today

Requested NRC Staff Positions on Fuel Qualification – Recap

Item 1.a: Confirm plans being implemented by the Advanced Gas Reactor Fuel Development and Qualification Program are generally acceptable and provide reasonable assurance of the capability of coated particle fuel to retain fission products in a controlled and predictable manner. Identify any additional information or testing needed to provide adequate assurance of this capability, if required.

Significance of Graphite Oxidation to Public Safety

- Graphite will chemically react (oxidize) with oxygen in air or in a helium-air gas mixture
- Nuclear grade graphite is much less reactive than other types of graphite due to its graphitized structure and high purity
- Oxidation of graphite is limited by
 - the amount of air in the helium gas mixture from the reactor building
 - the high flow resistance of the coolant channels to the core height ($L/D > 700$)
- Fuel particles are embedded in the graphite matrix within the fuel element
- Loss of all forced cooling and depressurization of HPB required for air to ingress
- MHTGR analyses for an assumed large HPB failure of 22 sq ft showed only 1% of core graphite oxidized after 30 days with 8 RB volumes of 100% air ingressed
 - Oxidation resulted in small contribution to heat generation compared to decay heat
 - Oxidation did not lead to loss of core geometry
 - No appreciable incremental radionuclide release due to oxidation
- NGNP analyses have shown that a break in the HPB leads to a small percentage of air in the gas mixture after the helium blowdown